UNITED STATES
NUCLEAR REGULATORYCOMMISSION
TECHNICAL TRAINING CENTER

BWR/4 TECHNOLOGY MANUAL (R-104B)

This manual is a text and reference document for the BWR/4 Technology Course. It should be used by students as a study guide during attendence at this course. This manual was compiled by staff members of the Technical Training Division in the Office for Analysis and Evaluation of Operational Data.

The information in this manual was developed or compiled for NRC personnel in support of internal training and qualification programs. No assumptions should be made as to its applicability for any other purpose. Information or statements contained in this manual should not be interpreted as setting official NRC policy. The data provided are not necessarily specific to any particular nuclear power plant, but can be considered to be representative of the vendor design.

A-23

R-104-B Course Outline

Day.	Subject/Chapter	Time
1	Training Center Introduction Course overview BWR types (1.4) Product Lines (1.5) Primary & auxiliary systems (1.7) including offgas Process control and inst system (1.8) Reactivity control (1.9) Containment systems overview (1.10) ECCS overview (1.11) Reactor Physics (1.12)	3.5
	Reactor Vessel and Vessel Inst. (2.1, 3.1)	2
	Fuel and Control Rods (2.2)	1
2	Control Rod Drive (2.3)	1
	Reactor Manual Control (7.1)	1
	Recirculation & RFC (2.4, 7.2)	2
	Main Steam & EHC (2.5, 3.2)	2
	Condensate & Feedwater (2.6)	1
3	Review	1
	FWCS(3.3)	1
	RCIC & RWCU (2.7, 2.8)	1.5
	NMS (5.0-5.5)	1.5
	RPS (7.3)	1.5
4	Review	1
	SLC (7.4)	1
	Containment Systems (4.0-4.4)	1.5
	ECCS (10.0-10.4)/Simulator Tour	3
5	Examination	3

STUDENT HANDOUT SHEET

TTC PHONE SYSTEM

1. Commercial: 423-855-6500
2 Incoming calls for students -- see paragraph on STUDENT MESSAGES
3. Classroom phones are connected on a common internal line and can only be used to call other areas inside the Training Center.
4. Wall phones in the 2nd floor student lounge area can be used for making outside calls.
5. To make local calls: dial 9 + local number
6. To make long distance calls: dial 9 + 1 + Area Code + Number

Note: *TTC is now on detailed billing for actual telephone usage and all calls are listed on a computer printout. Please limit calls home to no more than 5 minutes, per NRC Manual Chapter Appendix 1501, Part IV.D.5.*

AREA INFORMATION

1. Restaurants -- Eastgate Mall, Brainerd Road area
2. Hospital -- Humana in East Ridge -- Phone: 894-7870
3. Emergency Phone Number -- 911

COURSE RELATED ITEMS

1. Working hours are from 7:30 a.m. to 4:15 p.m. Classroom presentations are from 8:00 a.m. to 4:00 p.m. Lunch break will begin between 11:30 a.m. - 12:00 p.m. at the discretion of the instructor.
2. The Course Director and Course Instructor(s) are available to answer questions before and after class, during the breaks, and during lunch time with prior arrangement. Instructors not in the classroom can be reached via telephone. Please call ahead to ensure availability.
3. All course related materials (pencil, paper, manuals, notebooks, and markers) are provided. If there is a need for additional material or administrative service, please coordinate with the Course Instructors.
4. Shipping boxes will be provided for the mailing of course materials (manuals & notebooks). Each student must write their name and address to which the box is to be mailed on a mailing label and tape it to the outside of their box. The TTC staff will affix the proper postage.
5. Student registration for all TTC courses is accomplished through Training Coordinators. The TTC staff does not register students directly.

TTC SECURITY

NRC badges are required to be worn while at the TTC. Please promptly notify Course Director if your badge is lost or misplaced.

STUDENT MESSAGES

There is a printer located in the 2nd floor student computer room. All non-emergency student messages will be sent to this printer. It is your responsibility to check this printer for messages. If there are messages on the printer, please post them on the bulletin board above the printer.

COMPUTERS FOR STUDENT USE

There are several standard NRC NT Workstations located in the 2nd floor student computer room. These computers are equipped with the NRC suite of programs, including internet access. Instructions are posted for accessing individual e-mail accounts.

FIRST AID KITS

First Aid Kits are located in the instructor's desk of each simulator, in the second floor student lounge in the sink cabinet, and the sink cabinet in the staff lounge on the second floor. In addition, each location also has a "Body Fluid Barrier Kit". These kits are to be used in the event of personnel injury involving serious bleeding. Each kit contains two complete packets each with: 1 pair of latex gloves, 1 face shield, 1 mouth-to-mouth barrier, 1 protective garment, 2 antiseptic towelettes, and 1 biohazard disposable bag.

TAX EXEMPTION CERTIFICATES

NOTE: We do not have Tax Exempt Certificates for lodging in Chattanooga. Chattanooga is not one of the localities permitted to use these certificates. For a list of locations which are allowed to use them, see the Federal Travel Directory published monthly by GSA.

Please remember that you, as students, represent the NRC and when you knowingly avoid paying Tennessee State Tax, the results can have a negative effect on the Agency.

If you are not able to obtain adequate lodging and stay within the per diem rate established by GSA, advise your Management Support or DRMA office so the proper authorities can be notified.

FAX and COPYING AVAILABILITY

There are copy machines located on the 2nd, 3rd, and 4th floors. Students are asked to use the copiers on the 3rd and 4th floors for smaller jobs. If there is a need to copy larger jobs please coordinate with the Course Instructors for use of the large volume copy machine on the second floor.

Students needing to send a FAX can do so on the FAX machine located in the 2nd floor student computer room.

Students needing to receive a FAX during class time can have it sent to 423-855-6543. The FAX can then be picked up on the second floor in the Admin area.

STUDENT INFORMATION SHEET

PLEASE PRINT THE FOLLOWING INFORMATION:

Course Title: ____________________

Course Dates: ____________________

Name: ____________________
(How you want it to appear on Training Certificate

Social Security No: ___/___/_____ Job Title: ____________________

Phone No: __________ Mailing Address: ____________________

____
(No P.O. Boxes please)

Motel where you are staying: ____________________ Room No: __________

Name and number of person to call in an emergency: ____________________

____

Estimated Travel Cost (including transportation costs): ____________________

Name of Immediate Supervisor: ____________________

Name of Division Director: ____________________ **Name of Division:** __________

Please provide the following background information: (Please circle one)

1. Highest Level of Education:

   Doctorate Masters Bachelors Associate Other

2. Subject Matter Specialty:

   Engineering Physical Science Math or Statistics Other Science Other

3. Years of Nuclear Experience: >9 7-9 4-6 1-3 <1

4. Type of Nuclear Experience:

   Commercial BWR RO/SRO Navy Test Reactor Other
   Commercial PWR

5. Years with NRC: >9 7-9 4-6 1-3 <1

6. Previous TTC sponsored training attended: ____________________

____

____

Course Objectives
(R-104B)

The BWR/4 technology course is designed to provide the student with a general familiarity with the mechanical, instrumentation and control, and protective systems of the General Electric BWR/4 design. At the end of this course each student should have achieved a basic understanding of the following:

- Major process systems, purposes, normal system configuration, and safety related flowpaths.
- Major process instrumentation systems terminology, including selected interlocks and protection functions.
- PRA concerns in a change to the level of plant safety/risk as a result of system or component unavailability.

Course Outline

Day	Subject/Chapter Number	Time
1	Training Center Introduction Course Overview BWR types/Product lines (1.4 & 1.5) Primary & Auxiliary Systems, including Offgas (1.7) Process control and instrumentation systems (1.8) Reactivity Control (1.9) Containment systems overview (1.10) ECCS overview (1.11) Reactor Physics (1.12)	3.5
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	Fuel and Control Rods (2.2)	1
2	Control Rod Drive (2.3)	1
	Reactor Manual Control (7.1)	1
	Recirculation & Recirculation Flow Control (2.4 & 7.2)	2
	Main Steam & EHC (2.5 & 3.2)	2
	Condensate & Feedwater (2.6)	1
3	Review	1
	Feedwater Control System	1
	RCIC & RWCU (2.7 & 2.8)	1.5
	NMS (5.0 - 5.5)	1.5
	RPS (7.3)	1.5
4	Review	1
	SLC (7.4)	1
	Containment Systems (4.0 - 4.4)	1.5
	ECCS (10.0 - 10.4)	2
	Simulator Tour	1
5	Examination	2

TTC R-104-B
COURSE EVALUATION SHEET

I. Instructions:

In order to improve and maintain the quality and applicability of TTC courses it is necessary to obtain feedback from attending students. Please rate the following subject areas. Amplifying comments are desired but not required. Please place your amplifying comments in the section for written comments. Course evaluation should be identified by student to allow for follow-up and amplification of significant issues or suggestions.

II. Evaluation

	Strongly Disagree	Disagree	Agree	Strongly Agree
1. Stated course objectives were met.	______	______	______	______
2. Class materials were organized and presented in logical sequence.	______	______	______	______
3. Learning objectives were helpful in identifying important lecture concepts.	______	______	______	______
4. Classroom presentations adequately covered the learning objectives.	______	______	______	______
5. Course manual adequately covered course topics where applicable.	______	______	______	______
6. Course manual was organized so that it can be used as an effective study guide.	______	______	______	______
7. Course manual will be useful as a future reference.	______	______	______	______
8. Visual aids reinforced the presentation of course materials.	______	______	______	______
9. Completion of this course will assist me in my regulatory activities.	______	______	______	______

Signature: ______________________________

10. Overall course rating (considering merits of this course only):

Unsatisfactory	Marginal	Satisfactory	Good	Excellent
______	______	______	______	______

11. What did you like best or find most helpful about the course?

12. What did you like least about the course?

13. What subjects might be added or expanded?

14. What subjects might be deleted or discussed in less detail?

15. How will this course aid you in your ability to do your job as a regulator?

16. What could be done to make this course more useful in aiding you in your ability to effectively carryout your regulatory activities?

17. Additional comments.

Simulator Demonstration Instruction

A portion of this course may include demonstrations of system controls, indication, and operation on the BWR/4 simulator, located on the fourth floor of this building. The class may be split into two or more groups for these demonstrations. Due to the physical floor space limitations, it is important to stay in your assigned group.

Simulator demonstrations are planned demonstrations coordinated by the instructor. Students should **not** manipulate any switches on the simulator unless directed to by the instructor. The instructor console and computer area are not normally accessible areas for students. For hardware considerations, please do not place drinks on the simulator panels; ample desk area is available.

If the class is split into groups for simulator demonstrations, while one group is in the simulator, the other group or groups will be on self study. The self study time is to be used for reading manual chapters, review of material that has been presented, or if desired, review of material on an individual basis with an instructor. If instructor assistance is needed, please use the telephone in the classroom to call one of the course instructors. A telephone number list is attached beside the phone.

TABLE OF CONTENTS

1.0 INTRODUCTION

The GE Technology Manual is designed for use as a text for the GE Technology Course. The manual was written to reflect the BWR design. The numerical values and systems discussed in the manual are for a specific BWR. The reader should bear this in mind when attempting to use the manual as a general reference document.

1.0.1 Manual Organization

This manual has been organized to follow, as closely as possible, the order of the material presented in the above course. General subject areas are classified by chapters. Systems which fall under the general classification are arranged as sections within the chapter. Where applicable each section follows the same format; i.e., introduction, system description, component description, system features and interrelations, summary, and graphics.

The manual also has appendices containing definitions, symbols, abbreviations and a BWR Technology Course Syllabus, Outline and Purpose and Objective Sheets. Following completion of the instructors presentation and self-study, the student should be able to exhibit knowledge of these objectives.

1.0.1.1 System Introduction

The system introduction states the system purpose and functional classification. The purpose of the introduction is to orient the reader.

1.0.1.2 System Description

The system description provides the reader with an overview of the system and its components. Attention is focused on major components and their purposes without including the detail found in the component description.

1.0.1.3 Component Description

The components are listed in basic flow path order or block diagram arrangement. Each component is described in appropriate detail with specific set points and capacities often referenced in tables.

1.0.1.4 System Features and Interrelations

The system features and interrelations section includes such items as the operational features and limitations. It also identifies interfaces with other systems.

1.0.1.5 Summary

The summary is designed to key the reader to the major items contained in the chapter. It is important for the reader to recognize that the summary is not a substitute for a comprehensive review of the text material.

1.0.1.6 Graphics

The graphic package is located at the end of each section. The graphics are arranged to follow the text and are referenced in the written portion of the text.

1.0.2 Manual Use During Course Presentation

Proper use of the manual during class presentations can greatly aid the student in understanding the material presented. The student should follow the presentation using the figures and diagrams provided. Properly noting minor and major points on these figures should eliminate the necessity for taking comprehensive notes during the lecture.

1.1 COMMERCIAL NUCLEAR POWER PLANTS

To understand the BWR power plant, a basic knowledge of the major components and their functions is needed.

1.1.1 Nuclear Power Plants

A nuclear power plant is an arrangement of components and systems used to generate heat. The heat is used to make steam which is converted to electrical power. The principal components of a nuclear power plant are:

- nuclear fuel and moderator
- heat removal system
- control systems
- power conversion systems

1.1.2 Nuclear Fuel and Moderator

Nuclear fuel consists of a mixture of fissionable and fertile materials. The essential ingredient is a fissionable material, which is a material that readily undergoes nuclear fission when struck by neutrons. The only naturally available fissionable material is uranium-235 (U-235), an isotope (or form) of uranium constituting less than 1% of the element as found in nature. Two synthetic fissionable materials are plutonium-239 (Pu-239) and uranium-233 (U-233). When neutrons strike uranium-238 (U-238), which constitutes more than 99% of the natural uranium, Pu-239 is formed. For this reason U-238 is called a "fertile" material. The element thorium is also a fertile material, forming U-233 when struck by a neutron. The three basic fissionable materials may be used separately or with one of the fertile materials as fuel for a nuclear reactor.

The most commonly used fuel is uranium, either natural, or enriched in the U-235 isotope. Fuels may be solid or fluid and they may be used in different material forms: metals, alloys, oxides, or salts. A variety of solid fuel physicalshapes is used, including rods, plates, tubes, and other shapes, along with various methods for cladding (containing) the fuel. A moderator is a substance used in a reactor to slow down neutrons from high to low energy levels. Slowing down increases the probability of continued fission. Moderators commonly used include ordinary water, heavy water, and graphite. Liquid moderators can also serve as the coolant.

1.1.3 Heat Removal System

The heat removal system or cycle removes heat which is generated by the fission process in the reactor core. Heat removal system arrangements include single, double, and triple heat transfer cycles. An example of the single cycle system is the direct cycle boiling water reactor delivering steam to a turbine. Pressurized water reactors use two cycles, with the primary water transferring heat in a steam generator to produce steam for the turbine cycle.

1.1.4 Control Systems

In the general sense of the term, there are numerous control systems on modern reactors. The specific control system of concern here is reactivity control, which is the method by which the reactor core fission process is regulated. The basic method of accomplishing this regulation is to insert a neutron poisoning or absorbing material into the reactor core, thereby preventing those neutrons absorbed in the poison from causing fission in the fuel. There are other methods, some of which are specific to the BWR, which are discussed later in this text.

1.1.5 Power Conversion Systems

In modern reactor power plants, steam turbine generators are used to convert the energy of the steam into electrical power.

1.2 WATER COOLED REACTORS

Water is generally used as a coolant and a moderator for power reactors. Initially it was believed that water could not be permitted to boil in a reactor vessel because of the possibility of cladding burnout. This resulted in the early development of pressurized water reactors. The first pressurized water reactor went critical in 1953 at the AEC National Reactor Testing Station in Idaho

A different type of water cooled and water moderated reactor was started in 1953 with the first experiment to test the theory of boiling water in a reactor vessel and making steam directly. Successive experiments established the principle that boiling was not only acceptable but even advantageous for certain purposes.

It is only natural that water became the preferred primary reactor coolant. Reliability is a key factor and water has many important advantages that do not require extensive experimental programs. Water is a known quantity. It is cheap, and it was readily available when the reactor program was started. It has good heat transfer characteristics which can be extended beyond its normal narrow temperature range by pressurizing the water to inhibit boiling. Furthermore, water does not become significantly activated if kept pure. The induced radioactivity of the coolant is short lived so that maintenance is not hampered greatly.

The corrosive quality of water is known, and the pressurizing intensifies the corrosive action. An important inducement is that water serves as a moderator to slow down the neutrons; Its tendency to absorb neutrons can be overcome by enriching the fuel.

The disadvantages of using water as a moderator are: water must be highly pressurized to achieve reasonably high temperatures; pure hot water is highly corrosive and requires that the primary coolant system be constructed of special materials; water at high pressure and saturation temperature will flash to steam if the pressure is rapidly reduced, as in a rupture of the primary loop; and water can chemically react violently under certain temperature conditions with uranium, thorium, and structural materials.

The fundamental similarity in nuclear characteristics of water moderated reactors is determined basically by the nuclear and thermal properties of light water. Briefly, these similarities can be summarized as follows:

- Enriched fuel is required.
- Relatively low moderator to fuel ratios are employed.
- Relatively high excess reactivity is provided.
- Conversion ratios for existing types are low, but this is not an inherent characteristic.
- Power densities are comparatively high.

1.3 BOILING WATER REACTORS

In a boiling water reactor, the coolant is very pure water which boils adjacent to the fuel elements. The resulting steam-water mixture then proceeds to steam separators, where the water is separated from the steam bubbles. The water then goes back to the reactor core and the boiling operation is repeated. The steam which is formed passes from the steam separators, through the steam dryers, and to a turbine located outside the containment.

The major difference in the operating characteristics of a boiling water reactor core from other nuclear systems is a result of steam void production. Water affects both the heat generation and the neutron flux characteristics of a nuclear system because it serves the dual function of coolant and neutron moderator. If this water is allowed to boil, which greatly lowers the density of molecules, there is a significant change in nuclear performance. The boiling water reactor design results in a system that produces reactivity changes varying inversely with the steam void content in the core. This provides an inherent safety feature of the boiling water reactor; that is, a transient power increase will produce more steam voids, reducing reactivity, which reduces power and thus limits the excursion.

The fuel used in a boiling water reactor contains uranium in the form of an oxide. This eliminates the hazard involved in using uranium in metallic form. Moreover, before assembly into fuel elements, the uranium oxide is generally heated and converted into a ceramic material, somewhat like the bricks used to line fireplaces. This form of uranium oxide does not react chemically with the reactor coolant and does not burn in air.

1.4 TYPES OF BOILING WATER REACTORS

In direct cycle BWRs, as shown in Figure 1.4-1, steam leaving the reactor passes directly to the turbine. In an indirect cycle BWR, steam is passed to a primary coolant/secondary steam generator. No economic incentive exists for the latter cycle, although it does possess some advantage in that radioactive particles from the primary coolant normally cannot transfer to the steam used in the turbine generator portion of the plant. Dual cycle plants employ a combination of direct and indirect cycles. The first large utility owned BWR (Dresden 1) employed this dual cycle concept. Current BWRs use only the forced circulation direct cycle because it is more economical.

In a forced circulation direct cycle reactor system, the nuclear fuel generates heat within the reactor vessel and boils the water, producing wet steam that passes through internal steam separators and dryers. The water within the reactor is circulated through the core by two external recirculation pumps. The steam is directed from the reactor to the turbine, entering the turbine steam chest at about 950 psi and 540oF. Steam leaving the high pressure turbine passes through moisture separator units before being admitted to the low pressure turbines.

The low pressure turbines exhaust steam is condensed in the main condenser, which also provides system deaeration. The condenser is followed by a full flow demineralizer system through which all condensate and makeup must pass before entering the feedwater heaters.

The demineralizer system removes corrosion products produced in the turbine, condenser, and feedwater piping. It also protects the reactor against condenser tube leaks and removes other sources of impurities which may enter the system in the makeup water. The turbine cycle uses a conventional regenerative feedwater system. The feedwater temperature and the number of feedwater heaters are selected in accordance with normal power plant considerations of turbine cycle performance and economics.

1.4.1 Forced Circulation BWRs

Power density in a BWR core may be increased by using a mechanical pumping system to force the water through the core. This is called a forced circulation BWR. In this design a portion of the coolant in the annulus area between the core shroud and vessel wall is taken outside the vessel in recirculation loops, where it is increased in pressure by means of recirculation pumps. Water at increased pressure is pumped from the two recirculation loops back into the bottom of the reactor pressure vessel via jet pumps. Flow orificing of the fuel support pieces provides desired flow distributions. Water enters the core through the fuel assembly nosepieces and passes upward inside the channels containing the fuel bundles, where it is heated to become a two phase, steam-water mixture. The steam-water mixture leaves the top of the fuel assemblies and enters a plenum area above the core which directs the flow into the steam separators. Here the water is separated by centrifugal action. The rejected water is returned to recirculate through the pumping system. The steam then passes through a dryer where the last traces of water are removed. Dry steam exits through steam outlet nozzles at the top of the vessel body. Feedwater is added to the system through thermally sleeved spargers located in the downcomer annulus. Here the feedwater joins the water rejected by the steam separators before entering the recirculation pumping system.

1.4.1.1 Forced Circulation BWR Control Systems

The fluid flow rates and reactivity level in a forced circulation direct cycle BWR require rigid control of steam flow from the reactor, of feedwater flow into the reactor, of recirculation flow through the reactor, and of control rod position. The design of the control systems considers conventional power generation objectives, such as reliability, ease of operation, and response times of the controlling parameter. Beyond the traditional power generation objectives, the control systems must incorporate features specific to reactivity control and nuclear plant safety. These considerations involve effects on moderator temperature, fuel temperature, and void content as a function of steam pressure; steam generation and feedwater input; fuel exposure; and automatic shutdown of the nuclear chain reaction during unsafe or potentially unsafe conditions.

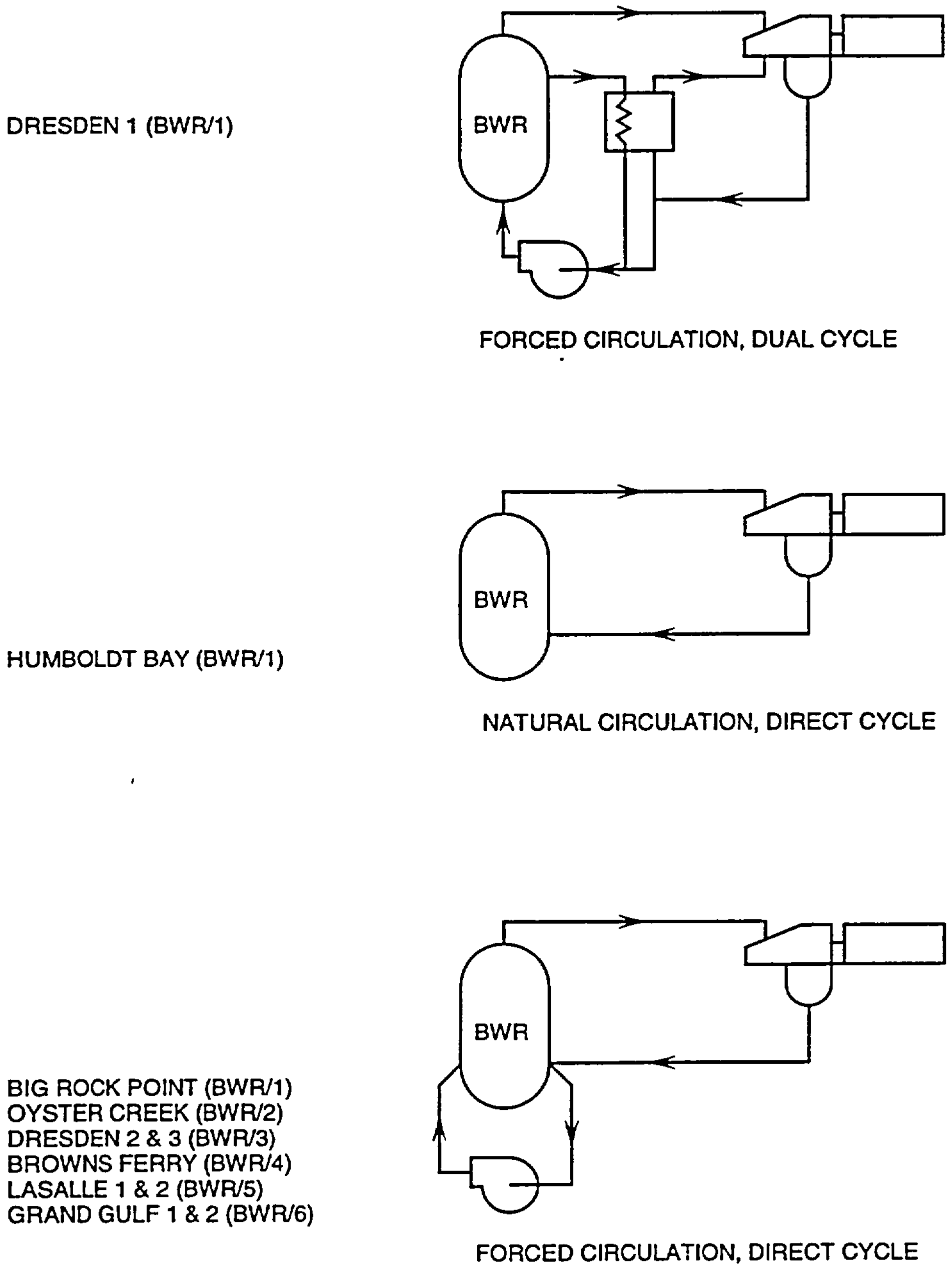

Figure 1.4-1 Boiling Water Reactors In Various Systems

1.05 Product Lines

Different product lines or classes of BWRs are designated numerically. There are currently six product lines, BWR/1, 2, 3, 4, 5, and 6. The following listing gives some basic information on the different product lines.

GE/BWR PRODUCT LINES

Product Line Number	Year of Introduction	Characteristic Plants
BWR/1	1955	Dresden 1, Big Rock Point, Humboldt Bay, KRB - Initial commercial BWRs - First internal steam separation
BWR/2	1963	Oyster creek - The first turnkey plant - Elimination of dual cycle
BWR/3	1965	Dresden 2 - The first jet pump application - Improved emergency core cooling system (ECCS)
BWR/4	1966	Browns Ferry - Increased power density 10%
BWR/5	1969	LaSalle - Improved Recirculation System performance - Improved ECCS performance - Mark II Containment
BWR/6	1972	Grand Gulf - Improved core performance - Improved rod control systems - Mark III Containment

This manual is written to the BWR/4 product line. Table 1.5-1 gives a listing of BWR/2 plants through BWR/6's along with some basic information on each plant.

Plant Name	BWR Class	Product Line	Containment *Type	MWT	MWE	Power Density KW/L	NO. Fuel Bundles	NO. Control Rods	IC or RCIC	LPCI or RHR *4	Feed Pump Type	Bypass Capacity	RFC Type	HPCI or HPCS	NO. Relief Valves	Press. Control Initial / Backup	*Type Offgas
Oyster Creek	2	63	Mark I	1930	620	33.6	560	137	IC	-	M	45	MG*5	FWCI	4	EPR/ MPR	AMB
Nine Mile Point #1	2	63	Mark I	1850	620	34.0	532	129	IC	-	M&T	45	MG*5	FWCI	6	EPR/ MPR	Low Temp
Dresden #2	3	65	Mark I	2527	809	41.08	724	177	IC	LPCI	M	45	MG	HPCI	5	EHC/ EHC	AMB
Millstone	3	65	Mark I	2011	690	40.08	580	145	IC	LPCI	MM	100*(3)	MG	FWCI	3	EPR/ MPR	Low Temp
Dresden #3	3	66	Mark I	2527	809	41.08	724	177	IC	LPCI	M	45	MG	HPCI	5	EHC/ EHC	AMB
Monticello	3	66	Mark I	1670	545	40.6	484	121	RCIC	RHR (A)	M	15	MG	HPCI	4	EPR/ MPR	Comp
Quad Cities 1/2	3	66	Mark I	2511	809	40.9	724	177	RCIC	RHR (A)	M	45	MG	HPCI	5	EHC/ EHC	AMB
Pilgrim	3	66	Mark I	1998	655	40.5	580	145	RCIC	RHR (A)	M	26	MG	HPCI	4	EPR/ MPR	AMB
Brown's Ferry 1/2/3	4	67	Mark I	3293	1065	50.7	764	185	RCIC	RHR (A)	T	26	MG	HPCI	13	EHC/ EHC	AMB
Vermont Yankee	4	67	Mark I	1593	514	51.0	368	89	RCIC	RHR (A)	M	100*(3)	MG	HPCI	4	EPR/ MPR	AMB
Duane Arnold	4	67	Mark I	1658	538	51.0	368	89	RCIC	RHR (A)	M	26	MG	HPCI	6	EHC/ EHC	AMB
Peach Bottom 2/3	4	67	Mark I	3293	1065	50.7	764	185	RCIC	RHR (A)	T	26	MG	HPCI	11	EHC/ EHC	Comp
Cooper	4	67	Mark I	2381	778	50.6	548	137	RCIC	RHR (A)	T	26	MG	HPCI	8	Digital (W)	Low Temp
Hatch 1/2	4	67	Mark I	2436	786	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB
Brunswick #1	4	67	Mark I (C)	2436	821	51.2	560	137	RCIC	RHR (A)	T	100*(3)	MG	HPCI	9	EHC/ EHC	AMB
Brunswick #2	4	67	Mark I (C)	2436	821	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB

Table 1.5-1 Plant Listing (Part 1)

Plant Name	BWR Class	Product Line	Containment *Type	MWT	MWE	Power Density KW/L	NO. Fuel Bundles	NO. Control Rods	IC or RCIC	LPCI or RHR *4	Feed Pump Type	Bypass Capacity	RFC Type	HPCI or HPCS	NO. Relief Valves	Press. Control Initial / Backup	*Type Offgas
Fitzpatrick	4	67	Mark I	2436	821	51.2	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB
Enrico Fermi	4	67	Mark I	3293	1093	50.0	764	185	RCIC	RHR (A)	T	26	MG	HPCI	11	EHC/ EHC	AMB
Hope Creek	4	67.5	Mark I	3293	1067	50.7	764	185	RCIC	RHR (C)	T	26	MG	HPCI	11	EHC/ EHC	AMB
Susquehanna 1/2	4	67.5	Mark II (C)	3293	1050	50.0	764	185	RCIC	RHR (A)	T	25	MG	HPCI	16	EHC/ EHC	AMB
Shoreham	4	67.5	Mark II (C)	2436	821	50.0	560	137	RCIC	RHR (A)	T	26	MG	HPCI	9	EHC/ EHC	AMB
Limerick 1/2	4	67.5	Mark II (C)	3293	1065	50.7	764	185	RCIC	RHR (C)	T	26	MG	HPCI	11	EHC/ EHC	AMB
LaSalle 1/2	5	69	Mark II (C)	3293	1078	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	11	EHC/ EHC	AMB
Hanford #2	5	69	Mark II	3323	1100	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	18	Digital (W)	Low Temp
Nine Mile Point #2	5	69	Mark II (C)	3323	1100	50.0	764	185	RCIC	RHR (B)	T	25	Valve	HPCS	18	EHC/ EHC	AMB
Grand Gulf	6	72	Mark III (C)	3835	1306	54.1	800	193	RCIC	RHR (B)	T	35	Valve	HPCS	20	EHC/ EHC	Low Temp
Perry	6	72	Mark III	3579	1200	56.0	748	177	RCIC	RHR (B)	M/T	35	Valve	HPCS	19	EHC/ EHC	Low Temp
Clinton	6	72	Mark III (C)	2894	995	52.4	624	145	RCIC	RHR (B)	M/T	35	Valve	HPCS	16	EHC/ EHC	Low Temp
River Bend	6	72	Mark III	2894	995	52.4	624	145	RCIC	RHR (B)	M	10	Valve	HPCS	16	EHC/ EHC	Low Temp

Notes: 1. Mark I : Drywell - Torus (Freestanding Steel Pressure Vessel)
Mark I (C) Drywell - Torus (Concrete with Steel Liner);
Mark II: Over/Under (Freestanding Steel Pressure Vessel);
Mark II (C): Over/Under (Concrete With Steel Liner);
Mark III Drywell - Containment (Freestanding Containment Vessel;
Mark III (C) Drywell - Containment (Concrete Containment With Steel Liner).

2. AMB Recombiner - with Charcoal Beds at Ambient Temperature
Low Temp Recombiner - with Charcoal Beds at Low Temperature(Zero degree f)
Comp Recombiner - with Comperssor and Storage Tanks for Extended Gas Holdup
3 Equipped with a Select Rod Insert Function
4. (A) Equipped with 2 RHR Loops, LPCI Mode Injects to the Recirculation System
(B) Equipped with 3 RHR Loops, LPCI Mode Injects Directly to the Vessel
(C) Equipped with 4 RHR Loops, LPCI Mode Injects Directly to the Vessel
5. Equipped with 5 Recirculation Loops and No Jet Pumps

Table 1.5-1 Plant Listing (Part 2)

1.06 PLANT LAYOUT

Modern BWR facilities are multiple unit plants. Greater economies can be realized with this arrangement by sharing certain functions within the facility. The principal buildings and structures associated with each unit of a particular site include the containment structure, the turbine building, the auxiliary building, the common control building, the diesel generator building, the standby service water cooling tower and basin, the enclosure (or shield) building, the common radwaste building, and the natural (or forced) draft cooling towers. A common structure is also provided which houses the administration offices, machine shop, and guardhouse. Location and orientation of the buildings of a site are shown in Figure 1.6-1.

1.06.1 Containment Building

The containment structure encloses the reactor coolant system, the drywell, suppression pool, upper pool, and some engineered safety feature systems and support systems.

1.06.2 Turbine Building

The turbine building houses all equipment associated with the main turbine generator as well as other auxiliary equipment.

1.06.3 Auxiliary Building

The auxiliary building is a structure that contains safety systems, fuel storage and shipping equipment and necessary auxiliary support systems.

1.06.4 Control Building

The control building is a multistoried structure which houses the main control room plus control and electrical systems required for safe operation of the plant.

1.06.5 Diesel Generator Building

The diesel generator building contains the emergency diesel generators and their associated equipment in individual rooms within the building.

1.06.6 Radwaste Building

The radwaste building houses various systems provided to process liquid, solid, and gaseous radioactive wastes generated by the plant.

1.06.7 Intake Structure

The intake structure houses the equipment providing the heat sink for the plant.

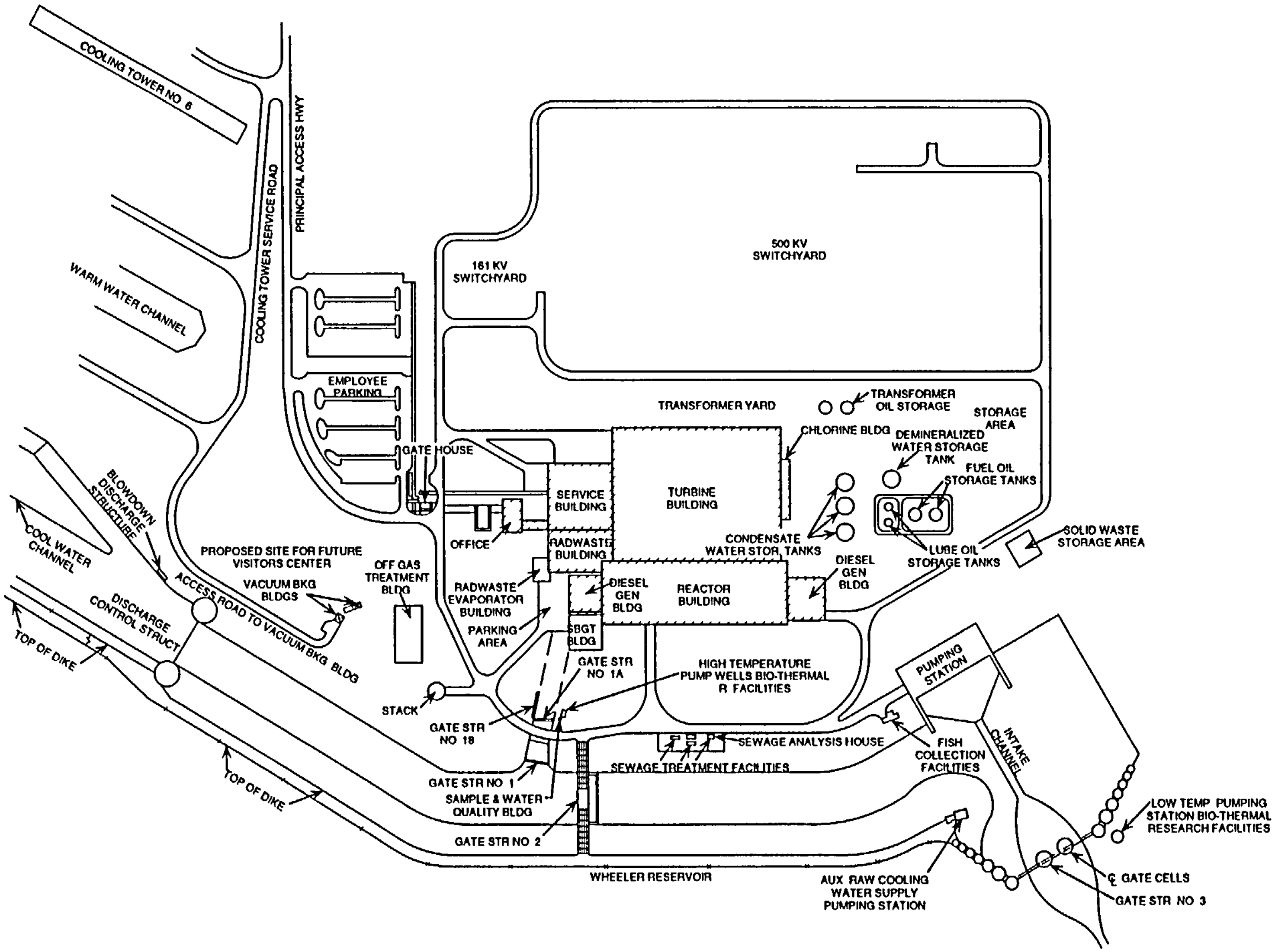

Figure 1.6-1 Site Plan and Building Arrangement

1.7 BWR PRIMARY AND AUXILIARY SYSTEMS

The BWR primary and auxiliary systems are the ones which are immediately involved in the direct cycle BWR concept as part of the steam cycle or else provide an auxiliary function for the direct cycle system. These systems are graphically displayed in Figure 1.7-1. The BWR direct steam cycle starts with the reactor vessel which is part of the reactor coolant pressure boundary and which contains the reactor core. The reactor core provides the heat source for steam generation and consists primarily of the nuclear fuel and control rods for regulating the fission process. The steam generated in the reactor vessel is routed to the steam loads and then condensed into water. The water is then purified, heated, and pumped back to the reactor vessel to again be heated. Water from the reactor vessel is circulated through external pumping loops and then returned to the reactor vessel to provide forced circulation of flow through the reactor core. Reactor water is continuously purified to minimize impurities. Should the reactor become isolated from its main heat sink, an auxiliary system automatically maintains the reactor core covered with water. The BWR primary and auxiliary systems are briefly discussed in the paragraphs that follow.

1.7.1 Reactor Vessel System

The Reactor Vessel System houses and supports the reactor core; provides water circulation to the reactor core to remove generated heat; separates the water and steam produced in the reactor core and delivers dry steam to the Main Steam System; and provides an internal, refloodable volume to assure core cooling capability following a loss of coolant accident (LOCA).

1.7.2 Fuel and Control Rods System

The fuel generates energy from the nuclear fission reaction to provide heat for steam generation.

The control rods control reactor power level, control reactor power distribution, and provide emergency shutdown capability.

1.7.3 Control Rod Drive System

The Control Rod Drive System positions control rods within the reactor core to change reactor power or to rapidly shutdown the reactor

1.7.4 Recirculation System

The Recirculation System provides forced circulation of water through the reactor core, permitting higher reactor power than with natural circulation.

1.7.5 Main Steam System

The Main Steam System directs steam from the reactor vessel to the main turbine and other steam loads and provides overpressure protection for the reactor coolant system.

1.7.6 Condensate and Feedwater System

The Condensate and Feedwater System condenses steam and collects drains in the main condenser, purifies, preheats, and pumps water from the main condenser to the reactor vessel.

1.7.7 Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System provides makeup water to the reactor vessel for core cooling when the main steam lines are isolated or when the Condensate and Feedwater System is not available.

1.7.8 Reactor Water Cleanup System

The Reactor Water Cleanup System maintains reactor water quality by filtration and ion exchange and provides a path for removal of reactor coolant when required.

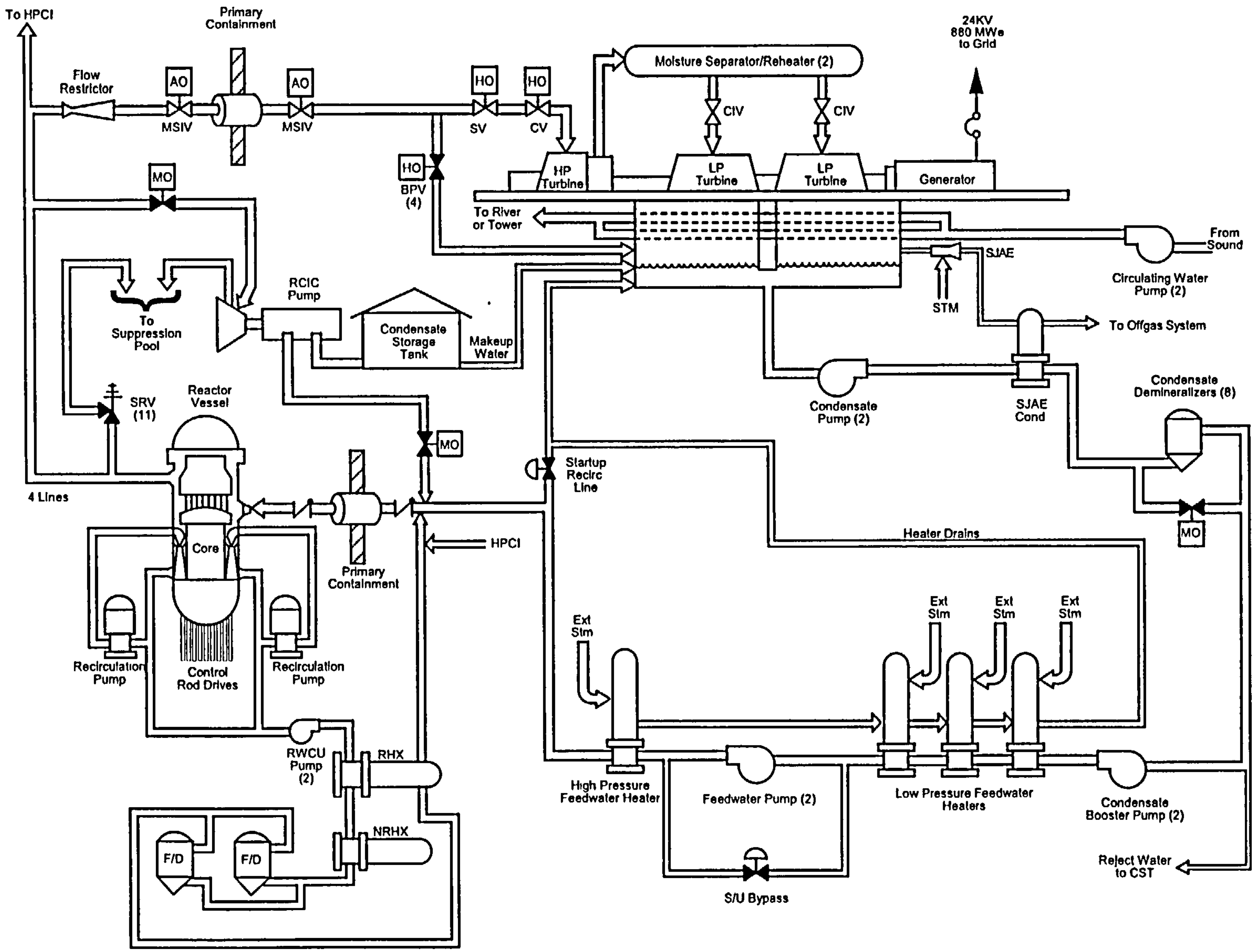

Figure 1.7-1 Simplified BWR Primary and Auxiliary Systems

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1.8 BWR PROCESS INSTRUMENTATION & CONTROL SYSTEMS

The systems discussed in this chapter are those that have to do with process instrumentation or process control. The control systems used to alter reactor core reactivity are discussed in Chapter 7. The process instrumentation and control systems are shown in Figure 1.8-1 and include the following systems: the Reactor Vessel Instrumentation System, the Electro Hydraulic Control System, and the Feedwater Control System.

1.8.1 Reactor Vessel Instrumentation System

The reactor vessel instrumentation System provides information concerning reactor vessel water level, reactor vessel pressure, reactor vessel temperature, and core flow rate. This information is used for control and automatic trip functions.

1.8.2 Electro Hydraulic Control System

The Electro Hydraulic Control System provides normal reactor pressure control by controlling steam flow in response to preset limits on operating parameters, such as desired turbine generator load and main steam pressure and provides the ability to conduct a plant cooldown.

1.8.3 Feedwater Control System

The Feedwater Control System controls the rate of feedwater flow to the reactor vessel to maintain the proper reactor vessel water level. The Feedwater Control System measures and uses total steam flow, total feedwater flow, and reactor vessel level signals to carry out its function.

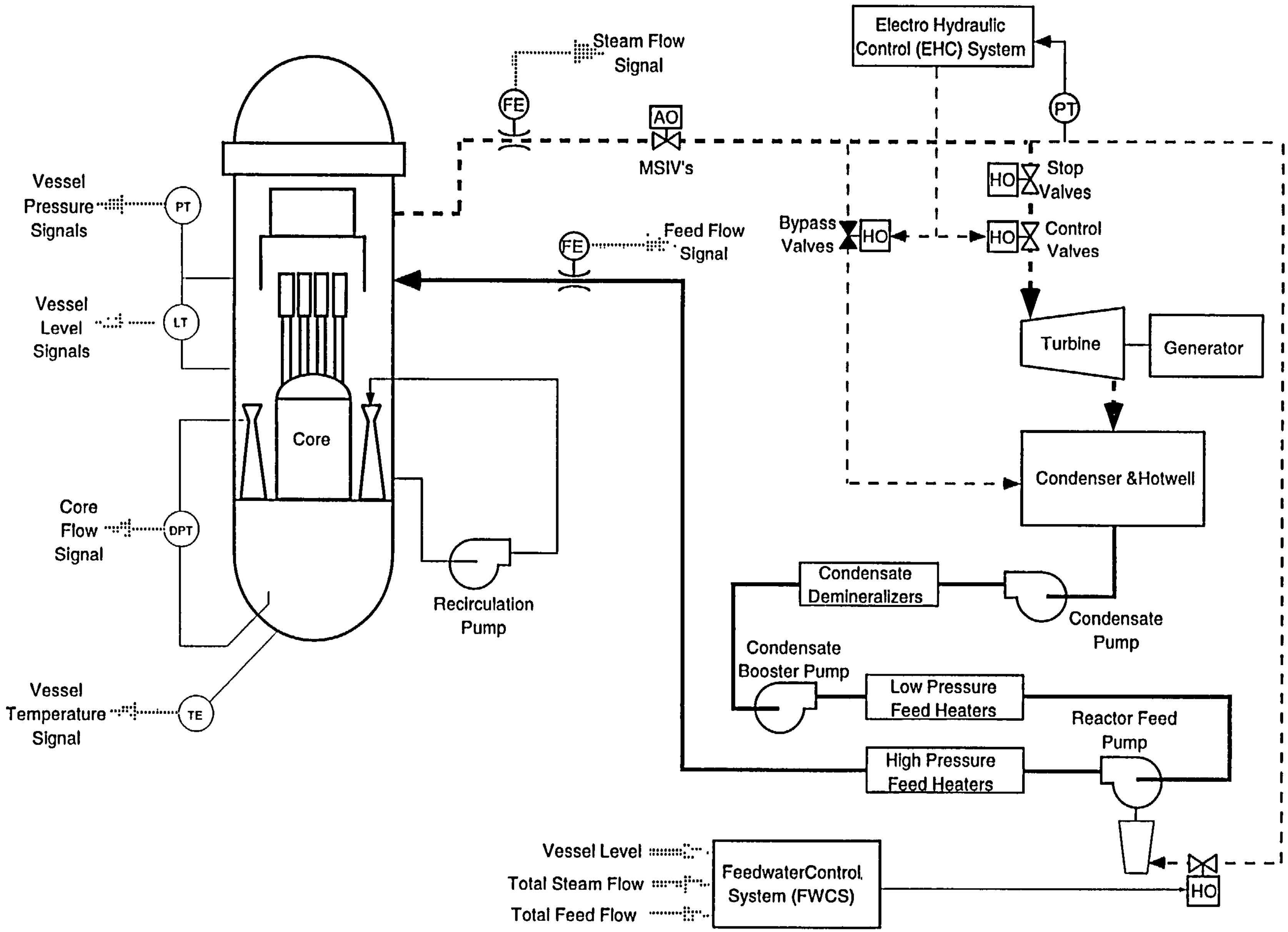

Figure 1.8-1 BWR Process Instrumentation & Control Systems

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1.9 REACTIVITY CONTROL SYSTEMS

The systems described in this chapter are the ones that are used to control the core reactivity under normal, abnormal, and emergency conditions. The reactivity control systems are shown in simplified form in Figure 1.9-1. The systems discussed in this section include the Reactor Manual Control System (RMC), the Recirculation Flow Control (RFC) System, the Reactor Protection System (RPS), and the Standby Liquid Control (SLC) System.

1.9.1 Reactor Manual Control System

The Reactor Manual Control System provides control signals to the Control Rod Drive System for normal control rod movement and prevents control rod movement during potentially unsafe conditions.

1.9.2 Recirculation Flow Control System

The Recirculation Flow Control system controls the means for rate of Recirculation System flow, allowing control of reactor power over a limited range.

1.9.3 Reactor Protection System

The Reactor Protection System initiates a reactor scram to preserve the integrity of the fuel cladding, to preserve the integrity of the reactor coolant system, and to minimize the energy which must be absorbed following a loss of coolant accident.

1.9.4 Standby Liquid Control System

The Standby Liquid Control system shuts down the reactor by chemical poisoning in the event of failure of the Control Rod Drive System.

1.9.5 Other Reactivity Control Features

There are two other features which can control core reactivity. Both of these are part of the Recirculation System. One feature is the ability to very the speed of the pump. This has the effect of increasing or decreasing recirculation loop flow rate. The other feature is the ability to trip the recirculation pumps which has the effect of rapidly reducing recirculation loop flow, core flow, and reactor power.

1.9.6 Other Parameters Affecting Reactivity

There are other parameters which effect core reactivity. The effects of fission product poison concentration changes are described in the reactor physics section. Two other parameters, reactor pressure and feedwater temperature, can also affect core reactivity. Reactor pressure is maintained at a constant value for a given reactor power level by the Electro Hydraulic Control (EHC) System. However, any transient change in reactor pressure causes steam voids in the core area to either collapse or be created causing a corresponding transient change in reactor power. Feedwater temperature is normally at some steady state value for a given reactor power level. Should the amount of feedwater heating change as the result of some transient, reactor power also changes unless compensated for by another reactivity control system.

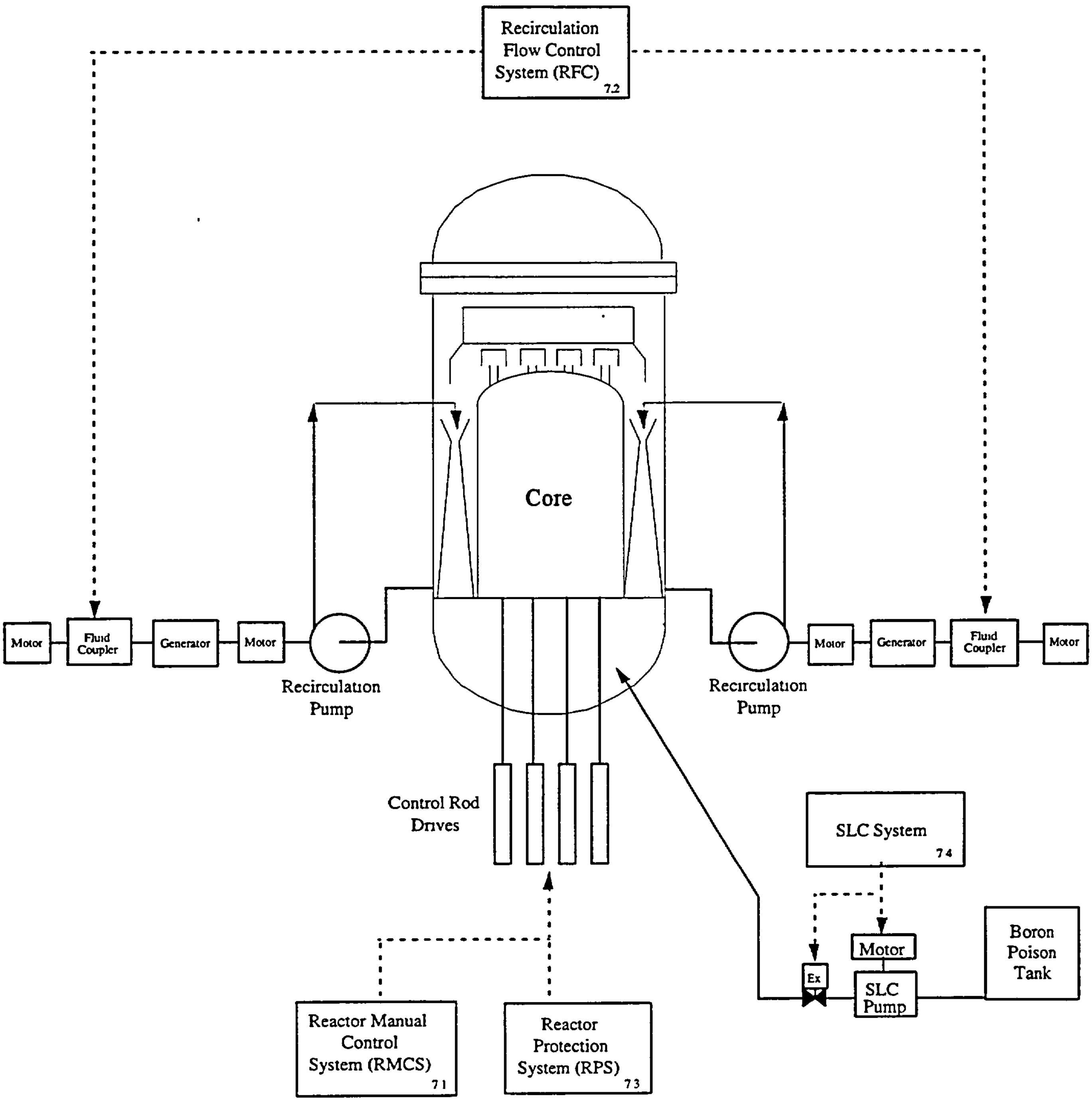

Figure 1.9-1 Simplified BWR Reactivity Control Systems

1.10 CONTAINMENT SYSTEMS

The BWR/4 product line uses the multibarrier, pressure suppression type of containment. The containment structure is similar to a standard dry containment and can be designed as either a free standing steel containment surrounded by a concrete shield building or as a concrete pressure vessel with a liner. The former design is referred to as the reference design while the latter is the alternate design. The reference design is discussed in this chapter.

The containment reference design, shown in Figure 1.10-1, uses a complex of structures to mitigate the consequences of postulated accidents. These structures are the:

- Primary Containment System
- Secondary Containment System

1.10.1 Primary Containment System

The Primary Containment System condenses steam and contains fission products released from a loss of coolant accident (LOCA) so that off site radiation doses specified in 10 CFR 100 are not exceeded. The Primary Containment System includes several auxiliary support systems.

1.10.2 Secondary Containment System

The Secondary Containment System minimizes the ground level release of radioactive material following an accident. It serves as a dilution and holdup volume for fission products which may leak from the primary containment following an accident.

1.10.3 Standby Gas Treatment System

The Standby Gas Treatment System processes exhaust air from the secondary containment boundary under accident conditions to limit radiation dose rates to less than the 10 CFR 100 guidelines. The Standby Gas Treatment System is also used to purge the drywell.

1.10.4 Primary Containment Isolation System

The Primary Containment Isolation System isolates the reactor vessel and various reactor plant systems which carry radioactive fluids or gases from the primary containment in order to prevent the release of radioactive materials to the environment in excess of the specified limits.

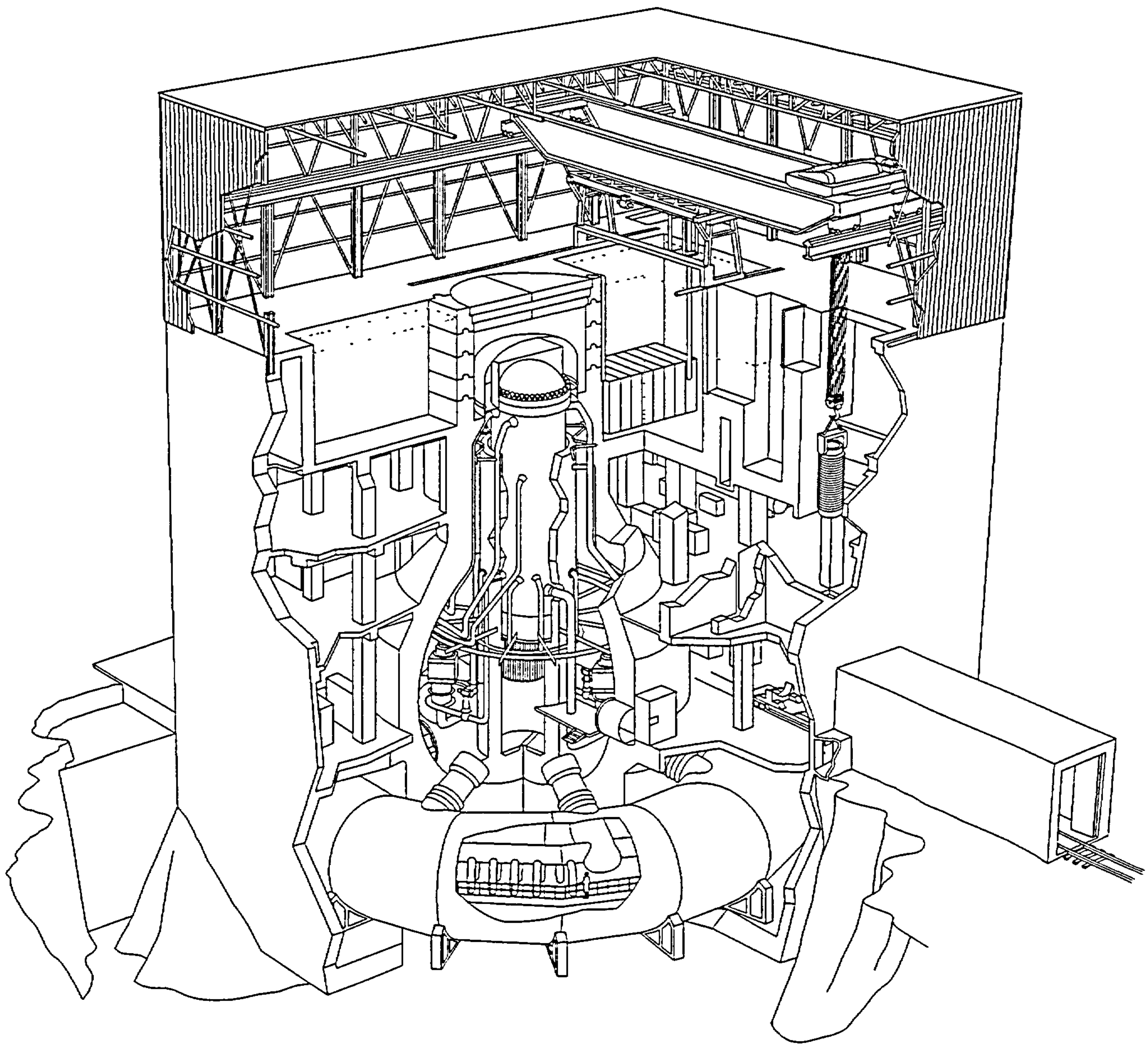

Figure 1.10-1 Mark I Containment

1.11 EMERGENCY CORE COOLING SYSTEMS

The purpose of the emergency core cooling systems (ECCS) is to provide core cooling under loss of coolant accident (LOCA) conditions to limit fuel cladding damage.

The ECCS, shown in Figure 1.11-1, consists of two high pressure systems and two low pressure systems. The high pressure systems are the High Pressure Core Spray (HPCS) System and the Automatic Depressurization System (ADS). The low pressure systems are the Low Pressure Core Spray (LPCS) System and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System.

1.11.1 High Pressure Core Spray System

The High Pressure Core Spray System maintains reactor vessel water inventory after small loss of coolant accidents (LOCAs), provides spray cooling for larger LOCAs, and backs up the function of the Reactor Core Isolation Cooling (RCIC) System under reactor vessel isolation conditions

1.11.2 Automatic Depressurization System

The Automatic Depressurization System (ADS) depressurizes the reactor vessel so that the low pressure emergency core cooling systems can inject water into the reactor vessel following small or intermediate sized LOCAs concurrent with HPCS System failure.

1.11.3 Low Pressure Core Spray System

The Low Pressure Core Spray (LPCS) System provides spray cooling to the reactor core to help mitigate the consequences of LOCAs when reactor pressure is low enough for the system to inject water into the reactor vessel.

1.11.4 Residual Heat Removal System

The low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System restores and maintains water level in the reactor vessel following LOCAs when reactor pressure is low enough for the system to inject water into the reactor vessel. The RHR System has several other operational modes, some of which are safety related and some of which are not.

1.11.5 Commission Requirements

The Code of Federal Regulations requires the ECCS to be designed so that following any LOCA the reactor core remains in a geometrical configuration amenable to cooling. The basic criteria are to limit fuel cladding temperature and oxidation to minimize clad fragmentation, and to minimize the hydrogen generation from clad oxidation to protect the containment.

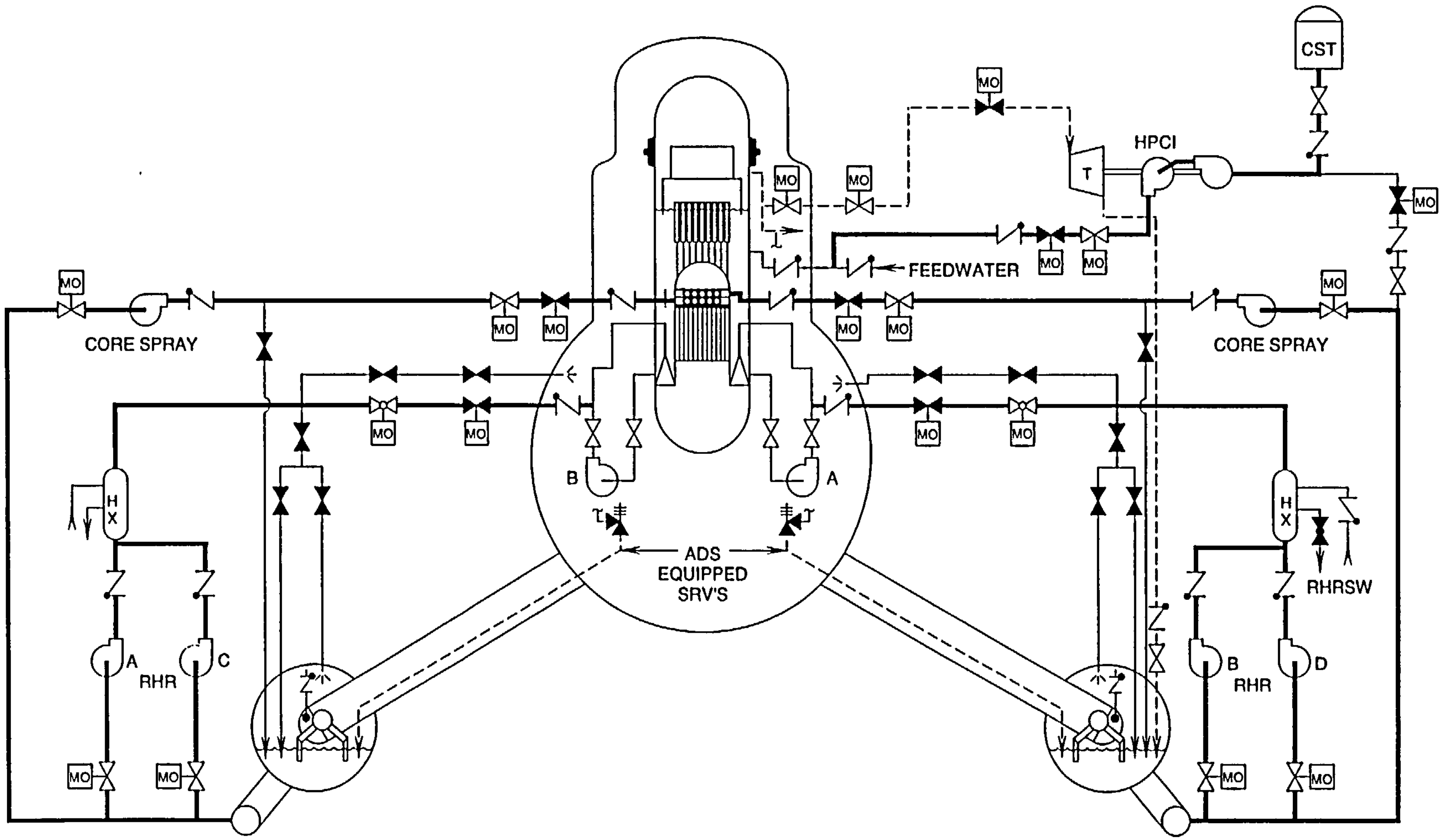

Figure 1.11-1 Emergency Core Cooling System

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1.12 REACTOR PHYSICS

The purpose of this section to provide a basic understanding of certain reactor physics concepts relating to Boiling Water Reactor (BWR) technology. For a more detailed discussion of this subject matter, additional reference material should be consulted.

1.12.1 Neutron Cycle

The principle of operating a nuclear reactor is based on neutron economy. Many of the processes within the reactor compete for the neutrons. The designer's problem becomes how to ensure that, under design conditions, each neutron in the previous generation produces one in the next generation.

A BWR is a thermal reactor. The term "thermal reactor" infers that fission is produced by neutrons in thermal equilibrium with the reactor core material. In a thermal reactor, the neutrons produced by thermal fission have very high energies and are called fast neutrons. They slow to thermal energies by collisions with moderator (water) nuclei. Some are then absorbed by fissionable nuclei with the subsequent fissions producing a new generation of neutrons. Figure 1.12-1 shows the neutron cycle for a thermal reactor and the process involved in slowing the neutrons down to thermal energies.

1.12.2 Effective Multiplication Factor

The ratio of neutrons in the new generation to the number in the previous generation is called the effective multiplication factor, Keff. Keff may be expressed as:

$$keff = \frac{\text{Neutrons produced}}{\text{Neutrons absorbed + neutron leakage}}$$

For a reactor to be critical the effective multiplication factor must be unity. Thus, there is a constant number of neutrons in each generation and the fission energy is released at a constant rate. When Keff is greater than unity the reactor is said to be supercritical and the power level will rise exponentially. Great care must be exercised that the rate of increase be kept within reasonable limits. When Keff is less than 1, the reactor is subcritical and there will be a decrease in neutron population and power.

1.12.3 Shutdown Margin

The shutdown margin is the amount by which the reactor is subcritical. Mathematically, this can be expressed as:

$$SDM = 1 - Keff \text{ (for keff} < 1)$$

The design shutdown margin will be specified for the plant, and tests will be conducted periodically to demonstrate that it is met. The value specified normally assumes that the strongest rod is stuck in the fully withdrawn condition. These tests demonstrate a margin of safety should that event occur.

1.12.4 Reactivity

Reactivity (symbol $\Delta K/K$) can be defined as the fractional change in neutron population per generation. Mathematically this can be expressed as follows

$$\Delta k/k = (keff-1)/keff$$

If a reactor is critical, $\Delta K/K = 0$ and keff = 1. If a reactor is subcritical, $\Delta K/K < 0$ and keff <1. If a reactor is supercritical, $\Delta K/K > 0$ and keff >1.

1.12.5 Reactivity Coefficients

The reactivity coefficients are merely a means of describing the effect on the multiplication factor (Keff) due to changes in a particular reactor parameter. They are usually expressed in terms of $\Delta K/K$/unit of parameter variable. There are three such coefficients at work in a BWR operating at power. These are the moderator temperature coefficient, the moderator void coefficient, and the fuel temperature coefficient.

1.12.5.1 Moderator Temperature Coefficient (αT)

The moderator temperature coefficient is defined as the change in reactivity caused by a 1°F change in moderator temperature. αT in a BWR is designed to be negative by establishing a proper moderator to fuel ratio. As the temperature of the moderator increases it becomes less dense. This decreases the amount of neutron moderation and increases the probability that a neutron will undergo nonfission absorption in a control rod or some core structural material.

1.12.5.2 Moderator Void Coefficient (αV)

The moderator void coefficient is defined as the change in reactivity caused by a one percent change in void concentration in the core. αV is negative because an increase in moderator voids causes the leakage components of Keff to be dominant as with the moderator temperature coefficient.

1.12.5.3 Fuel Temperature Coefficient (αD)

The fuel temperature coefficient, also referred to as the doppler coefficient, is defined as the change in reactivity caused by a 1°F change in fuel temperature. The effect of αD is attributed to the fact that U238 and Pu240 have an affinity to capture neutrons of certain energy levels, called resonance peaks. Capturing the neutron is a form of neutron leakage since the neutron cannot cause fission. As the fuel is heated, the resonance absorption of neutrons in U238 and Pu240 increases, causing a decrease in core reactivity. Hence αD is also negative.

1.12.5.4 Reactivity Coefficient Values

Approximate numerical values for the three reactivity coefficients are as follows:

αV - 1 X 10-3 $\Delta K/K$ /% voids

αT - 1 X 10-4 $\Delta K/K$/°F moderator

αD - 1 X 10-5 $\Delta K/K$ /°F fuel

From these values it is easy to see that the void coefficient is dominant when the reactor is in the power range (i.e. a substantial percentage of voids).

1.12.6 Reactor Control

When a reactor is operating at steady state, as discussed in section 1.12.2, the effective mulitplication factor, Keff is unity; there is a constant number of neutrons in each generation and the fission energy is released at a constant rate. To change the power level of the reactor, the rate of fission energy released or the number of neutrons in each generation must be changed. While the power is increasing or decreasing, Keff differs from unity. The operator must limit the changes in Keff (reactivity) so that the reactor can be controlled.

An expression for how reactor power will vary with time is

$$P = P_o e^{t/T}$$

where P is the power of some time, t ; Po is the power at time t = o; and T, the reactor period, is the time for the power to change by a factor of the natural log base (e).

For the reactor to be at steady state P = Po and, therefore T = ∞. Reactor period is a very important concept and is one of the most responsive indicators of reactor conditions: All reactors employ an automatic safety system which will rapidly shutdown (scram) the reactor if the period gets too short, avoiding dangerous power excursions.

The following sections apply the principles covered above to explain reactor control during a normal startup from a cold shutdown condition. The response of the reactor can be broken down into three distinctly different areas: the source range, the intermediate range and the power range. Each of these areas will be covered below.

1.12.6.1 Source Range

The source range covers approximately 10-8 %to 10-4% reactor power. Neutron flux is in this range when the reactor is shutdown and during the initial phases of a reactor startup. When starting the reactor, the power is controlled by control rod withdrawal to establish the reactor in the critical condition.

Before withdrawing any control rods to begin the startup, the operator verifies that there is a minimum count rate indicated on the source range monitor (Section 5.1). Source neutrons are present in the reactor so that it is possible to see the approach to critical on the reactor instrumentation. If there were no source neutrons present, the instrument range would not be sensitive enough to detect a positive, increasing period until the power was high enough to indicate on the instrumentation. By the time the power was indicating on the instruments, the period could be very short and a startup accident could occur.

As the operator withdraws control rods, a nonfission absorber is removed from the core causing Keff to increase and neutron multiplication can be seen on the instrumentation. When enough control rods have been withdrawn to raise Keff to 1, the reactor is critical. Further control rod withdrawal then establishes a rate of power increase into the intermediate range to commence a plant heat up.

1.12.6.2 Intermediate Range

The intermediate range encompasses power levels from approximately 10-5% to about 40%. The power level is controlled by control rods and the negative feedback effect of the moderator temperature coefficient of reactivity.

Recall that the intermediate range is entered slightly supercritical. To the unwary operator this can be a problem. A rapidly increasing flux level can give a high level flux scram from the intermediate range monitors (Section 5.2). Attention must be paid to proper range switching. When power increases to the point of adding heat, moderator temperature will increase. As moderator temperature increases, the density of the moderator decreases. The decrease in moderator density adds negative reactivity to the core which causes reactor power to turn and start to decrease. The operator now withdraws a control rod (or rods) until core reactivity is positive and the process is repeated. The rate at which this is done and the magnitude of positive reactivity controls reactor power, which in turn controls the heatup rate. This process is continued until the plant is at its operating temperature. The purpose of power control here is to control the heat up rate of the plant to prevent undue thermal stresses on plant structural materials.

1.12.6.3 Power Range

For this discussion, the power range is considered to be a power level greater than 1%. The reactor still responds by the P = Poet/T expression in the power range. However, this response is difficult to recognize because it is impossible to establish a stable period. There is always some factor resisting a power increase or decrease. Therefore, an attempt to establish a positive period by control rod withdrawal is immediately terminated at some fractionally higher power because of increased voids, fuel temperature, or moderator temperature. Thus, the negative reactivity coefficients related to the fuel temperature and core void fraction provide a negative feedback to power changes. The effect of the moderator temperature coefficient (aT) is limited here because of the pressure and temperature relationship at saturation, and the constant pressure maintained by the Electro-Hydraulic Control System.

From 1% to approximately 25%, reactor power is controlled to establish plant conditions required for rolling the turbine and picking up initial load on the generator. From 25% to 100%, reactor power is controlled to control the generator load. Reactor control is accomplished through use of control rods and the recirculation system. When the plant startup is complete, power range operation is normally steady state. Small power changes required to accommodate grid load changes are normally done by adjusting recirculation flow. Longer term reactivity changes, caused by fission product poisons (hours) and fuel depletion (weeks), are offset by control rod movement under the direction of the nuclear engineer.

1.12.7 Fission Product Effects

During the course of operation of a nuclear reactor, the fission fragments and their many decay products accumulate. Among these substances there are some, xenon-135 and samarium-149 in particular, which have large probabilities (cross sections) for thermal neutron absorption. These nuclei, therefore, act as reactor poisons and effect the multiplication factor, chiefly by decreasing the thermal utilization.

The concentration of fission product poisons in a reactor is related to the thermal-neutron flux (or density). Consequently, when power is changed, so that there is an accompanying change in the neutron density, the concentration of fission product poisons will be affected and this will, in turn, influence reactivity. This effect must be considered in core design and control system design.

Because of more rapid formation and removal, and higher absorption probabilities, xenon presents a greater short term reactivity control problem than samarium.

1.12.7.1 Xenon

Xenon-135 can be formed both directly and indirectly in fission and can be removed by radioactive decay and by neutron capture. As a result of the two opposing types of reactions, the con- centration of xenon-135 will reach an equilibrium value while the reactor operates at a specific power level.

Neglecting the direct production by fission, the formation and loss of xenon-135 can be represented schematically as:

<u>burnout</u>

+n

$$ {}^{135}_{53}\mathrm{I} \xrightarrow[\text{6.7 hr } t_{1/2}]{\text{formation}} {}^{135}_{54}\mathrm{Xe} $$

decay

9.2 hr t1/2

From this diagram, one can associate the formation term largely with prior operation (needed to produce iodine-135) and the burnout term +n with current power level (produces neutrons for capture by xenon-135). For example, after some hours of operation, if the reactor is suddenly shutdown, the burnout term is removed and the xenon simply cannot decay as fast (9.2 hr $t^{1/2}$) as it is being formed from iodine (6.7 hr $t^{1/2}$) and for a while, the xenon concentration will increase until most of the iodine has decayed and the xenon decay dominates. Conversely, a sudden rise in power level equates to an increase in neutron flux (the burn-out term) and, for a while, the xenon depletes faster than it can be formed from decay of the iodine backlog created at the lower power level.

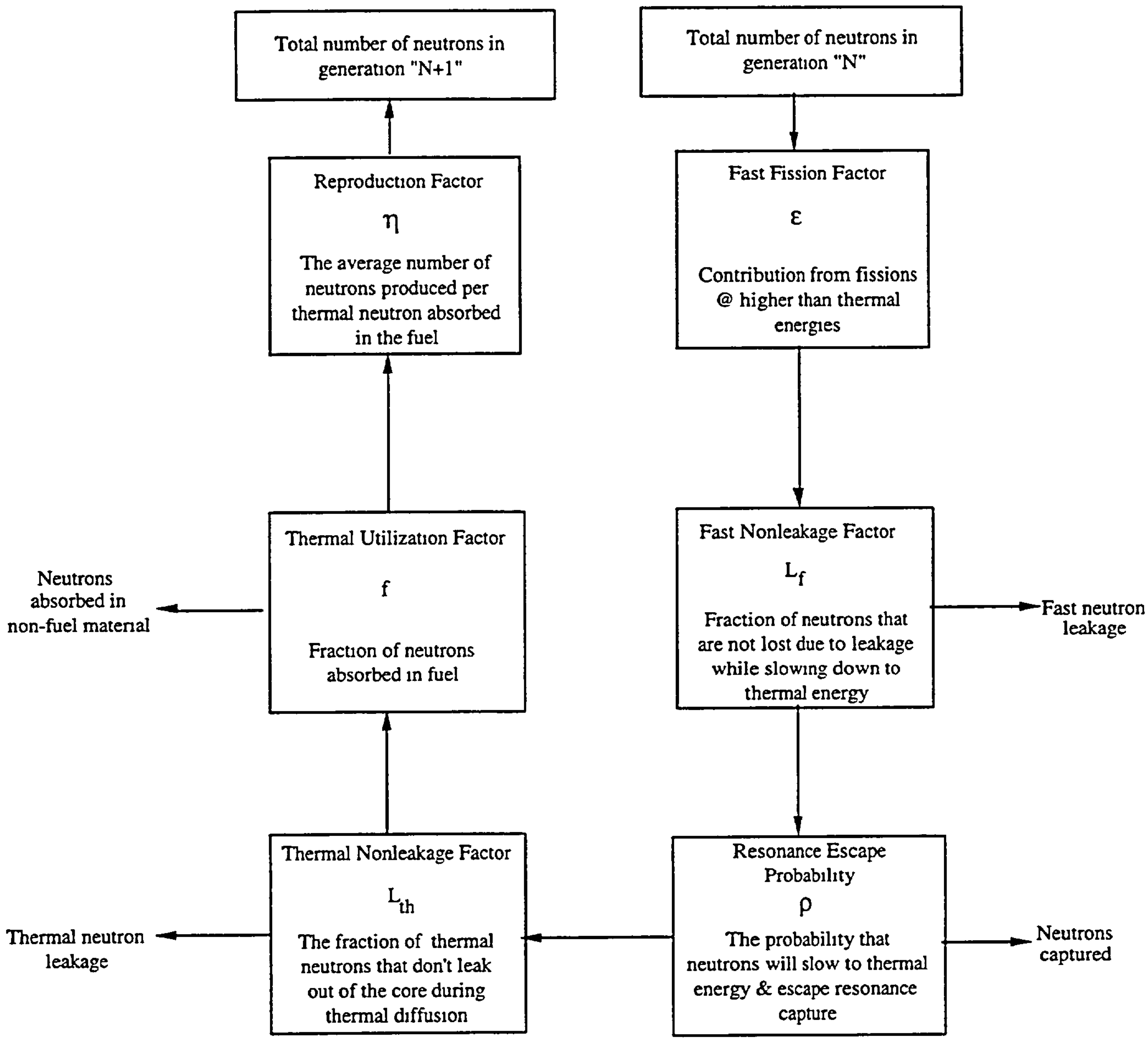
Operationally, as xenon changes, other poisons in the core (control material such as control rods) must be added or removed to maintain criticality. Whenever the power level is changed, the operator must consider the effect xenon will have on continued operation.

1.12.7.2 Samarium

Next to xenon-135, the most important fission product poison is samarium-149. It is the end product of the decay chain

$$ {}^{149}_{60}\mathrm{Nd} \xrightarrow[1.7\ \mathrm{hr}\ t^{1/2}]{} {}^{149}_{61}\mathrm{Pm} \xrightarrow[47\ \mathrm{hr}\ t^{1/2}]{} {}^{149}_{62}\mathrm{Sm} $$

Samarium-149 is a stable nuclide, is removed only by burnout and has a probability for neutron absorption about 100 times smaller than that of xenon-135. Considering on operating BWR, it takes several days for samarium to build up to an equilibrium concentration and its reactivity effects following power changes are several magnitudes less than xenon's. Samarium is often treated as an equilibrium poison and its transient effects neglected.



$$K_{eff} = \varepsilon \, L_f \, \rho \, L_{th} \, f \, \eta = \frac{\text{Total \# neutrons in "N+1" generation}}{\text{Total \# neutrons in "N" generation}}$$

When critical : $K_{eff} = 1$

Figrue 1.12-1 Neutron Cycle in a Thermal Reactor

1.13 THERMAL LIMITS

The purpose of having thermal limits is to minimize the radiological release from the plant during normal operation, abnormal operation, abnormal operational transients, and postulated accidents by restricting plant operation so that the fuel cladding integrity is maintained.

Fuel clad damage (loss of cladding integrity) is defined, for design purposes, as a perforation of the cladding which permits release of fission products. The mechanisims which could cause fuel damage are:

- Rupture of the cladding due to strain caused by relative expansion of the uranium dioxide pellet and the fuel cladding.
- Severe overheating of the fuel cladding caused by inadequate cooling.

1.13.1 Thermal Limit Description

Thermal limits are provided for normal operation and transient events to maintain the integrity of the fuel cladding. This objective is achieved by limiting fuel rod power density to avoid overstressing the fuel cladding because of fuel pellet- cladding differential expansion and by maintaining nucleate boiling around the fuel rods so that the transition to film boiling is avoided. The thermal limits established for these purposes are the linear heat generation rate (LHGR) limit and the minimum critical power ratio (MCPR) limit.

A thermal limit is provided for postulated accidents to maintain the core geometry by minimizing the gross fuel cladding failure because of the heatup following a loss of coolant accident (LOCA). The thermal limit established for this purpose is the maximum average planar linear heat generation rate (MAPLHGR) limit.

The basic thermal limits are shown in Figure 1.13-1.

1.13.2 Background Information

In order to understand the BWR thermal limits, it is necessary to have an understanding of related background material such as heat transfer and fluid flow characteristics. This subject material is discussed in the paragraphs that follow.

1.13.2.1 Heat Transfer

In light water reactor operation, heat is transferred from the fuel center line to the light water moderator which comes into contact with the outer fuel cladding surface. The heat can be transferred by conduction, convection, or radiation. Conduction and convection are the modes of heat transfer of primary interest in nuclear power plant operations.

1.13.2.1.1 Conduction

When heat is applied to a material, the kinetic energy of the atoms or molecules of the material is increased. Because of this increase in kinetic energy, the particles have a greater tendency to collide with each other. When these collisions occur, the particles transmit a portion of their kinetic energy to neighboring atoms. This is conduction.

This is the process by which heat generated in the fuel pellet is transmitted to the outer clad surface. In relation to the fuel rod, this conduction flow is in a horizontal plane from the fuel center line to the cladding surface.

1.13.2.1.2 Convection

Convection is the process of transmitting heat from a heated surface or area to a fluid by circulation or mixing of the fluid. Convection takes place only in fluids.

The application in this case deals with a fluid (moderator) flowing past a metallic fuel cladding surface. Fluids have a tendency to adhere to solid surfaces resulting in the formation of a stagnant film on the surface. This film is normally very thin and heat is transferred across this film by a combination of conduction and convection. After the heat penetrates the film, it is transferred rapidly through the remainder of the fluid by mixing.

The resistance of heat flow is so low that there is virtually no temperature variation through the bulk of the fluid (moderator) at any given elevation along the fuel rod.

1.13.2.1.3 Radiation

Radiation heat transfer is the transmission of heat in the form of radiant energy from one object to another across an intervening space. This form of heat transfer is avoided in nuclear power plants because very high temperature differentials are required to transfer a significant amount of heat. These high temperatures would cause degradation of materials if allowed to occur.

1.13.3 Boiling Heat Transfer

The boiling heat transfer curve is shown in Figure 1.13-2. The amount of heat transferred from the fuel cladding to the coolant is greatly affected by the coolant properties and by the thermal and hydraulic conditions of the coolant. The rate at which heat is transferred from the cladding, the heat flux, is dependent on the specific temperature difference between the cladding and the coolant (DT) and the heat transfer coefficient. The heat flux may be plotted against the temperature difference between the cladding and the coolant. This curve can be divided into boiling regions corresponding to the regions shown in Figure 1.13-2. The heat transfer coefficient in each region is controlled by the mode of heat transfer in that region.

The first region is single phase convection heat transfer. The heat flux increases somewhat with increased DT.

The second region is associated with subcooled nucleate boiling. Subcooled nucleate boiling is boiling at the cladding surface with the bulk coolant temperature not yet at saturation temperature. The steam bubbles may collapse before departing from the cladding surface or they will collapse as they enter the subcooled region after departing from the surface. This mode of convective heat transfer is a complicated mixture of single phase convection and nucleate boiling modes of heat transfer.

The next region is fully developed bulk nucleate boiling. Nucleate boiling is a very efficient mode of heat transfer because of high turbulence created by the boiling process. Nucleate boiling is maintained in the core in all modes of normal operation and in all transient conditions caused by a single operator error or equipment malfunction.

The heat flux increases as the temperature difference between the cladding and the coolant increases. There is a point, however, where the heat transfer coefficient no longer increases with an increased DT. There is a transition boiling regime where the boiling mode changes from nucleate boiling to film boiling. This region is highly unstable and is characterized by the intermittent physical rewetting of the heated surface by the coolant. The beginning of this region is called onset of transition boiling (OTB) and is labeled so on Figure 1.13-2. The OTB point is avoided in the BWR by remaining within the critical power ratio thermal limit.

The crosshatched region represents temperature oscillations which take place during transition boiling. At a given heat flux the clad surface temperature will oscillate between a point on the right in the crosshatched region and a point on the left in the crosshatched region along a horizontal line. This is caused by intermittent physical rewetting of the clad surface. At the point of onset of transition boiling (OTB), the temperature oscillations reach 25°F in magnitude; this is defined as the critical power.

1.13.3.1 Fluid Flow

Figure 1.13-3 shows the different flow patterns which can exist in a fuel bundle during normal operation. As coolant (single phase liquid) enters the fuel bundle it is slightly subcooled, and begins to gain heat from the forced flow convection mechanism. Because of subcooling, there is little or no bubble formation. As energy is gained, the coolant temperature increases until nucleate boiling with its attendant bubble formation begins Early states of nucleate boiling occur while the bulk coolant in the bundle is below liquid saturation enthalpy, and the bubbles readily collapse as the turbulent flow and their buoyancy sweeps them away from the clad surface. A point will be reached where the bulk coolant enthalpy is at liquid saturation, (bulk boiling) and the bubbles will no longer collapse in the coolant as they are swept away. The bubbles now begin to exist separately throughout the bulk coolant, causing a significant steam fraction to be present in the coolant. From this point to the bundle outlet, the bubbles continue to form at the fuel rod surface (nucleate boiling) and be swept into the coolant and begin to coalesce into larger and larger slugs of steam (slug flow). At the outlet of very highest powered fuel bundles, steam may fill most of the bundle flow area between fuel rods, but a thin annulus of water adheres to the fuel rod surfaces (annular flow). In this annular flow region, the wetted rod surface is still transferring heat through the nucleate boiling mechanism.

1.13.3.1.1 Fuel Channel Parameter Characteristics

Figure 1.13-4 shows a plot of coolant and fuel bundle temperature versus flow path length of an average fuel bundle. Coolant enters the bottom of the core, flows upward around the fuel rods, and absorbs energy from heat transfer originating from the nuclear process. Because of the peculiar characteristics of neutron caused fission reactions, the average heat flux (Q/A) produced from fission in the core assumes a shape somewhat like that shown in Figure 1.13-4. The highest heat flux is in the core interior, hence some fuel bundles have a higher than average heat flux while some have lower than average.

The coolant temperature curve rises as heat is added, until temperature saturation occurs, and coolant bulk boiling begins. From this point the coolant temperature remains constant all the way to the core outlet. Because the coolant is changing phase, the coolant temperature profile is not altogether descriptive of coolant energy increase. A better description is obtained by plotting coolant enthalpy change, which continuously increases from core inlet to outlet, with the largest rate of increase at the maximum value of heat flux.

The curve for fuel rod surface temperature rises and then levels at a constant value above coolant temperature. The initial rise is caused by the DT across the film required to accommodate the heat flux (Q/A) during single phase forced convection heat transfer. The point where the fuel rod temperature levels off is because of the start of nucleate boiling. Nucleate boiling is an excellent heat transfer mode; therefore, even though the heat flux increases, the DT across the boiling film remains relatively constant.

The curve for fuel rod center line temperature is above that of the fuel surface temperature. The amount that the center line temperature is greater than surface temperature depends directly on the heat flux. The beneficial effects of nucleate boiling on center line temperature can also be seen. As long as nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This, in turn, keeps the fuel center line temperature at a lower value than if single phase convection were the mode of heat transfer from surface to liquid

1.13.3.1.2 Fuel Temperature Profile

Figure 1.13-5 graphically illustrates a typical fuel temperature cross section with nucleate boiling at a high heat flux and its benefits. As long as nucleate boiling is occurring on the fuel rod surface, the fuel rod surface temperature is only slightly greater than liquid temperature. This, in turn, keeps the fuel center line temperature at a lower value than if single phase convection were the mode of heat transfer from surface to liquid.

1.13.4 The LHGR Limit

Linear heat generation rate (LHGR) is the surface heat flux integrated over each square centimeter of cladding material in one linear foot of a fuel rod. Limits on LHGR are set to limit the strain on the fuel clad because of relative expansion of the fuel pellets and the clad. A value of 1 percent plastic strain of the cladding is conservatively defined as a threshold below which fuel damage due to fuel clad overstraining is not expected to occur.

Limits on LHGR are set to limit the strain on the fuel clad because of relative expansion of UO^2 and the clad. Relative expansion arises from several sources: the UO^2 fuel thermal expansion coefficient is approximately twice that of zircaloy; the fuel pellets operate at higher temperatures than the clad; the fuel pellets undergo irradiation growth as they are exposed; and the fuel pellets crack and redistribute toward the clad due to thermal stress. Cladding cracking because of differential expansion of pellet and clad is prevented by limiting fuel pin power such that 1% plastic strain does not occur.

The linear heat generation rate required to cause 1 percent cladding strain in 8x8 fuel is approximately 25 kW/ft for unirradiated fuel, but decreases with burnup to a value of approximately 20 kW/ft at a local exposure of 40,000 Mwd/MT.

The design LHGR for 8x8 fuel is 13.4 kW/ft, which provides a margin to the 1% plastic strain threshold. The maximum LHGR in the core is monitored by calculating peaking factors.

1.13.5 The APLHGR Limit

Average planar linear heat generation rate (APLHGR) is the average LHGR of all fuel rods in a given fuel bundle in a given horizontal plane (actually a 6 inch slab or node). This parameter is important in the core heatup analysis for a loss of coolant accident (LOCA).

In the event of a LOCA, the heat stored in the fuel at the time of the accident and the decay heat produced following the accident could significantly damage the fuel. In the case of some LOCA's, nucleate boiling is maintained around the fuel long enough for the majority of the stored energy in the fuel to be conducted to the coolant; thus fuel damage is minimized. In the design basis LOCA, however, the core region is voided of liquid coolant in a relatively short time. If the fuel is operating at a high enough power level, the stored energy in the fuel could lead to a gross cladding failure and possible severe degradation of core geometry. Once water is removed from the cladding, radiation is the only heat transfer mechanism.

Gross cladding failure is prevented by placing a limit on the power level which would result in a peak cladding temperature (PCT) of 2200°F following a LOCA. The thermal limit specified is the APLHGR which is used because the PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod less than or equal to the design LHGR corrected for fuel densification.

The thermal limit in this case is given in terms of the incore maximum APLHGR (the MAPLHGR) and is specified for each individual fuel type as a function of fuel exposure. The units of the MAPLHGR are the same as those of the LHGR (kW/ft) even though the parameters are different.

1.13.6 The CPR Safety Limit

Critical power is the fuel bundle power required to cause transition boiling somewhere in the fuel bundle. The critical power ratio (CPR) of a fuel bundle is the ratio of its critical power to the actual fuel bundle operating power. The CPR is a measure of how close to transition boiling a fuel bundle is operating. The minimum value of the CPR for all fuel bundles in the core is the minimum critical power ratio (MCPR) and represents the fuel bundle which is the closest to transition boiling. MCPR limits are imposed to avoid fuel damage due to severe overheating of the clad.

The MCPR safety limit is set at 1.07 to ensure that in the event of an abnormal operating transient, more than 99.9% of the fuel rods in the core are expected to avoid transition boiling. The margin between a MCPR of 1.0 (onset of transition boiling) and the safety limit is derived from a detailed statistical analysis of uncertainties in monitoring the core operating state and in the boiling transition correlation.

1.13.7 Summary

In summary, there are three basic thermal limits; linear heat generation rate (LHGR) and minimum critical power ratio (MCPR) are limited to ensure fuel clad integrity during normal and transient operation, maximum average planar linear heat generation rate (MAPLHGR) is limited to meet ECCS criteria. LHGR limits are imposed to prevent fuel cladding perforation because of mechanical stress of the fuel pellets. MCPR limits are imposed to prevent cladding perforation because of the onset of transition boiling (breakdown of the heat transfer mechanism), and are modified to account for transients and flow conditions less than rated. MAPLHGR limits are imposed to restrict the amount of stored energy in the fuel thus limiting the rate of cladding heatup on a LOCA.

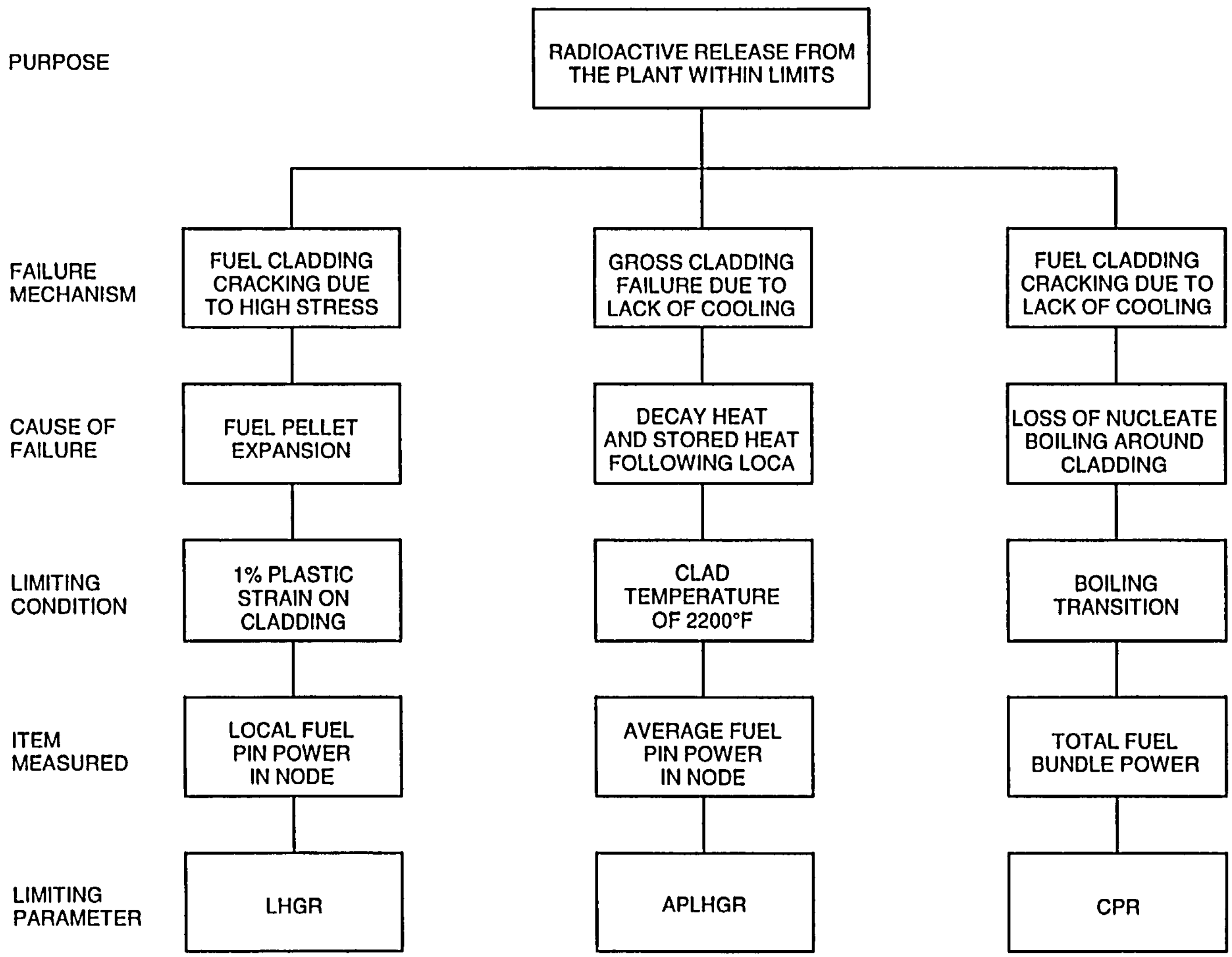


FIGURE 1.13-1 THERMAL LIMITS

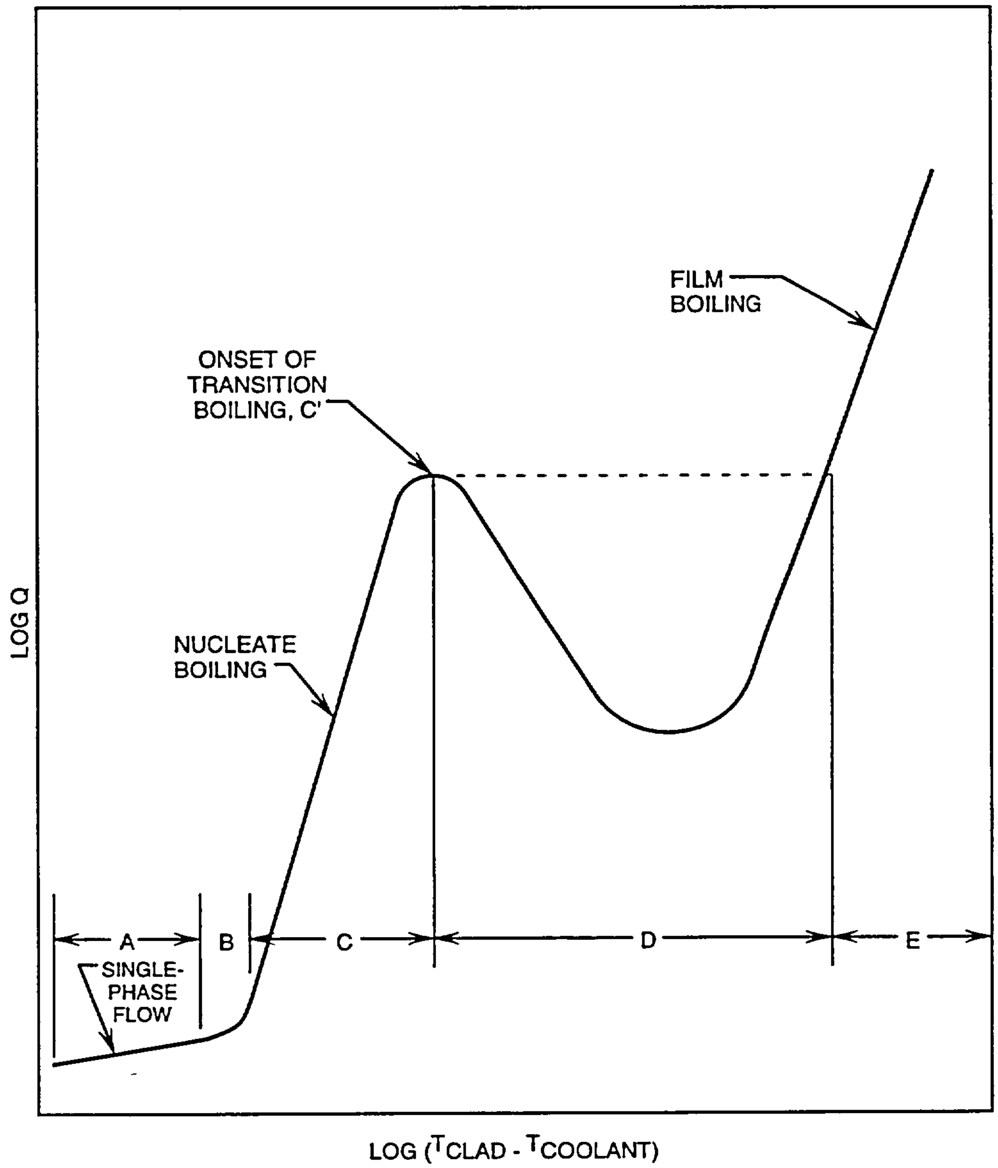
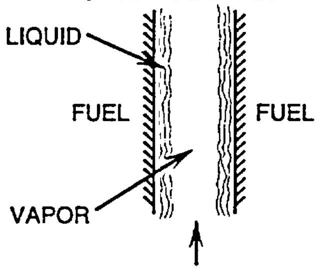


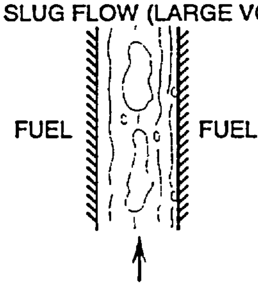
FIGURE 1.13-2 HEAT FLUX VERSUS TEMPERATURE DIFFERENCE BETWEEN CLADDING AND COOLANT

e. ANNULAR FLOW



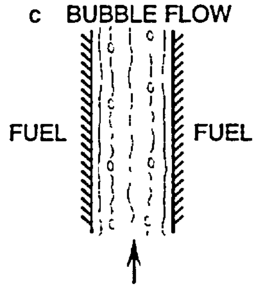
1) VAPOR FORMS A CONTINUOUS PHASE IN THE SPACE BETWEEN FUEL ELEMENTS (SATURATED VAPOR)
2) SLOWER MOVING LIQUID TRAVELS ALONG THE FUEL ELEMENT SURFACE
3) EXPERIENCED NEAR CHANNEL EXIT IN HIGH POWER CHANNELS

d SLUG FLOW (LARGE VOIDS)



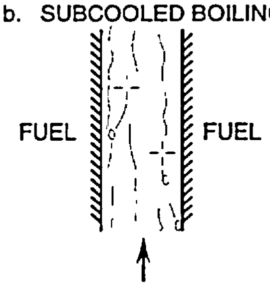
1) BUBBLES ARE FORMING ON WALL AND COALESCING TO FORM VAPOR "SLUGS" IN MIDCHANNEL
2) SIGNIFICANT VOID FRACTION
3) IN HIGH POWER CHANNELS SLUG FLOW BEGINS BELOW CORE MIDPLANE

c BUBBLE FLOW



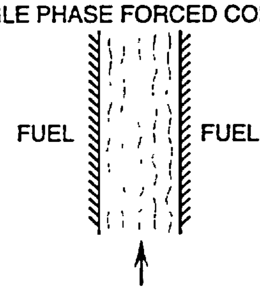
1) BUBBLES FORM ON THE WALL BUT DO NOT COLLAPSE IN COOLANT STREAM, I.E., "BULK BOILING"
2) BULK COOLANT TEMPERATURE IS AT SATURATION TEMPERATURE
3) BUBBLES ARE NOT YET COALESCING, LOW QUALITY

b. SUBCOOLED BOILING



1) BUBBLES FORM ON THE WALL BUT COLLAPSE IN THE BULK COOLANT STREAM
2) NO NET VAPOR GENERATION
3) A VERY SMALL VOID FRACTION
4) OCCURS ABOVE THE FORCED CONVECTION REGION

a. SINGLE PHASE FORCED CONVECTION



1) NO BUBBLES ARE FORMED, THE COOLANT REMAINS IN A SINGLE PHASE (LIQUID)
2) OCCURS AT VERY BOTTOM OF COOLANT CHANNEL

FIGURE 1.13-3 FUEL CHANNEL BOILING CONDITIONS

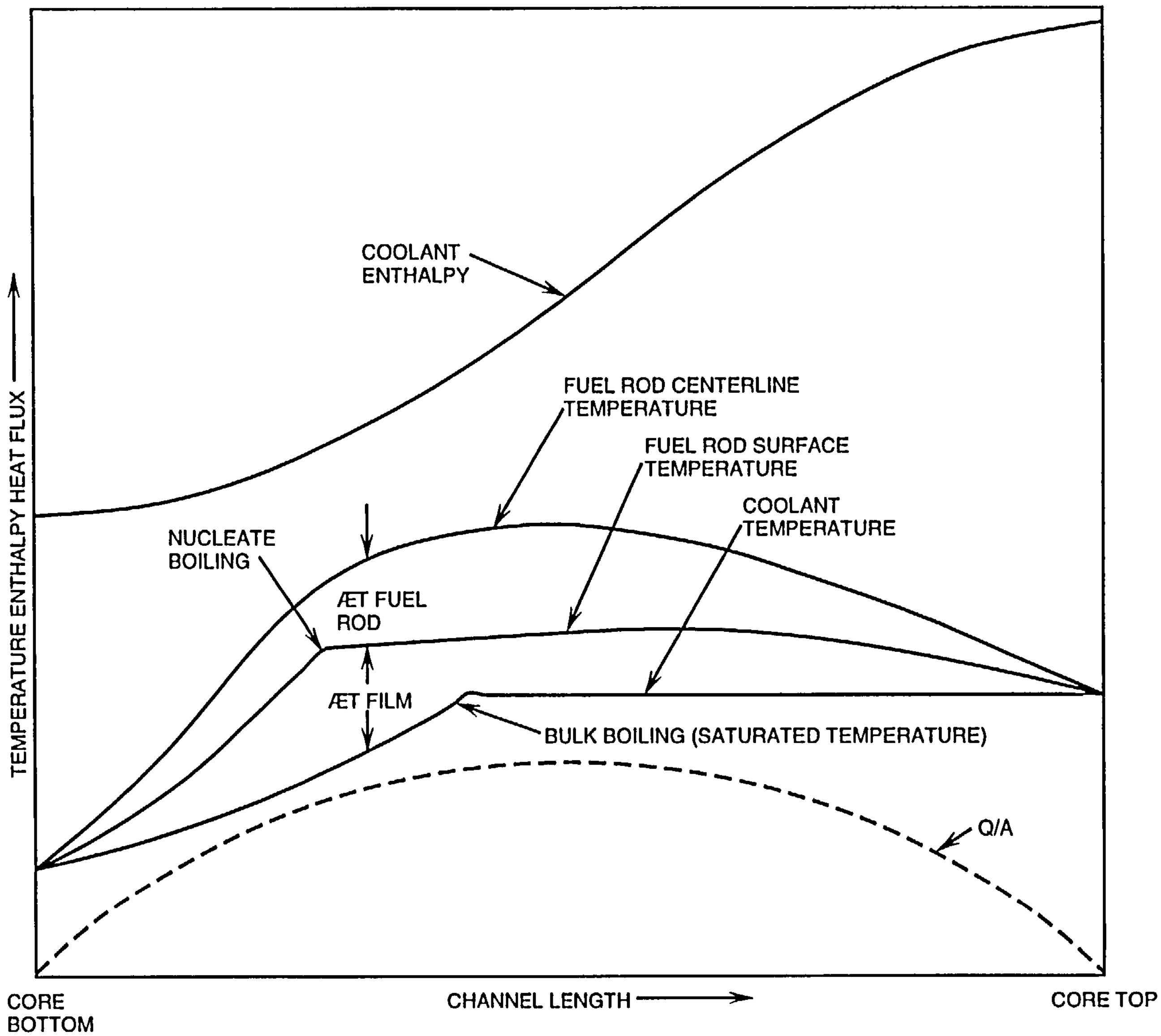


FIGURE 1.13-4 PLOT OF COOLANT AND FUEL BUNDLE TEMPERATURE VS FLOW PATH LENGTH

0102

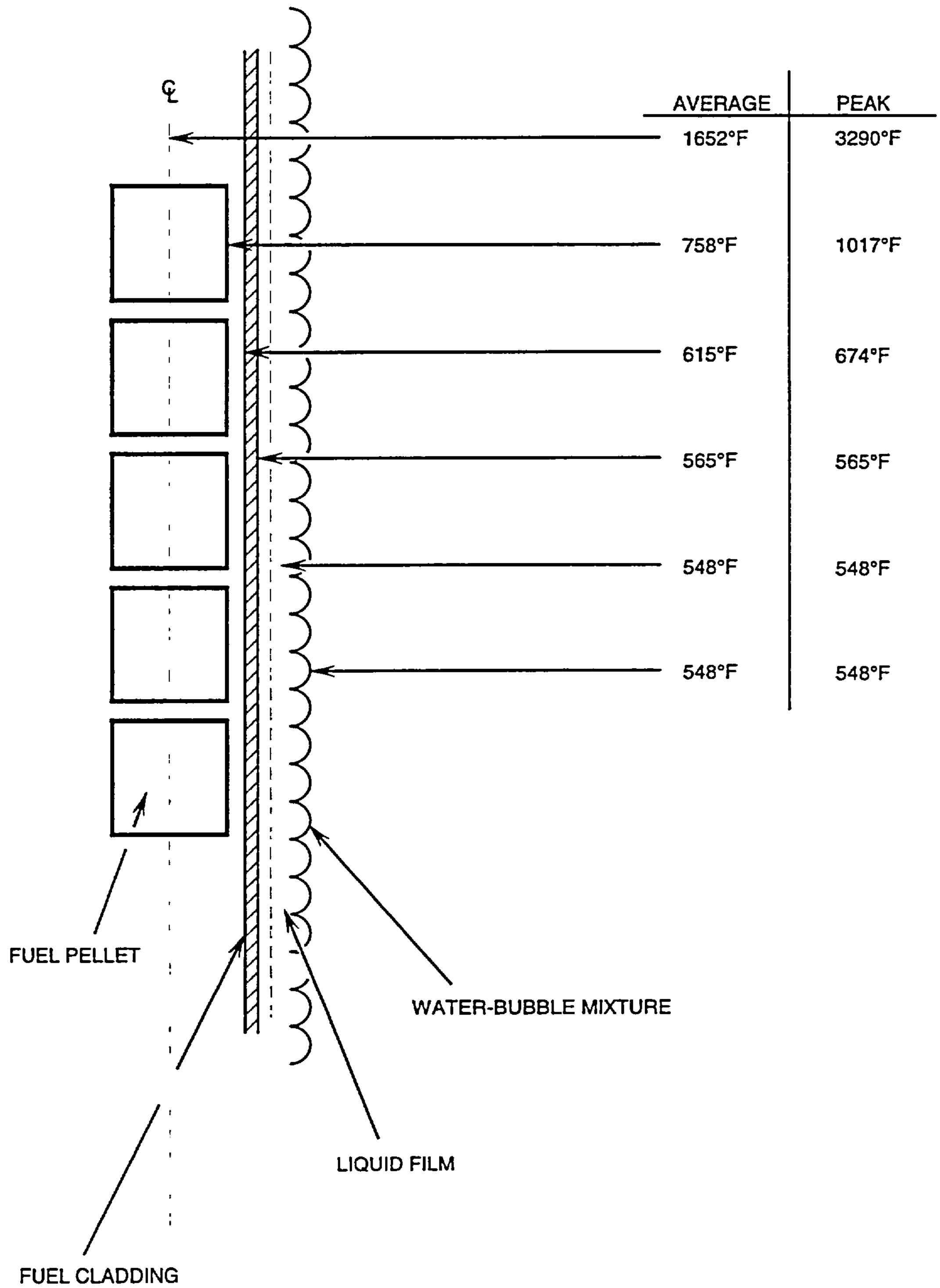


FIGURE 1.13-5 FUEL TEMPERATURE CROSS SECTION (8 X 8 FUEL)

2.0 PRIMARY AND AUXILIARY SYSTEMS

The BWR primary and auxiliary systems are the ones which are immediately involved in the direct cycle BWR concept as part of the steam cycle or else provide an auxiliary function for the direct cycle system. These systems are graphically displayed in Figure 2.0-1. The BWR direct steam cycle starts with the reactor vessel which is part of the reactor coolant pressure boundary and which contains the reactor core. The reactor core provides the heat source for steam generation and consists primarily of the nuclear fuel and control rods for regulating the fission process. The steam generated in the reactor vessel is routed to the steam loads and then condensed into water. The water is then purified, heated, and pumped back to the reactor vessel to again be heated. Water from the reactor vessel is circulated through external pumping loops and then returned to the reactor vessel to provide forced circulation of flow through the reactor core. Reactor water is continuously purified to minimize impurities. Should the reactor become isolated from its main heat sink, an auxiliary system automatically maintains the reactor core covered with water. A reactor heat balance is provided on Figure 2.0-2. The BWR primary and auxiliary systems are briefly discussed in the paragraphs which follow.

2.0.1 Reactor Vessel System (Section 2.1)

The Reactor Vessel System houses the reactor core, serves as part of the reactor coolant pressure boundary, supports and aligns the fuel and control rods, provides a flow path for the circulation of coolant past the fuel, removes moisture from the steam exiting the reactor vessel, provides an internal floodable volume to allow for reflooding the reactor core following a loss of coolant accident, and limits downward control rod motion following a postulated failure of control rod drive housing.

2.0.2 Fuel and Control Rods System (Section 2.2)

The fuel generates energy from the nuclear fission reaction to provide heat for steam generation. The control rods control reactor power level, control axial and radial power (neutron flux) distribution to optimize core performance, and provide adequate excess negative reactivity to shutdown the reactor from any normal operating or accident condition at the most reactive time in core life.

2.0.3 Control Rod Drive System (Section 2.3)

The Control Rod Drive System makes gross changes in core reactivity by positioning the neutron absorbing control rods in response to Reactor Manual Control System (RMCS) signals and rapidly inserts all control rods to shut down the reactor in response to Reactor Protection System (RPS) signals.

2.0.4 Recirculation System (Section 2.4)

The Recirculation System provides forced circulation of water through the reactor core, thereby allowing a higher power level to be achieved than with natural circulation alone.

2.0.5 Main Steam System (Section 2.5)

The Main Steam System directs steam from the reactor vessel to the turbine generator, bypass valves, reactor feed pump turbines, and other selected balance of plant loads; directs steam to certain safety related systems under abnormal conditions; and provides overpressure protection for the reactor coolant pressure boundary.

2.0.6 Condensate and Feedwater System (Section 2.6)

The Condensate and Feedwater System condenses turbine exhaust or bypass steam, removes impurities, heats the feedwater and delivers the water back to the reactor vessel at the required rate to maintain correct inventory. The feedwater piping also provides a means for the Reactor Water Cleanup (RWCU) System, the Reactor Core Isolation Cooling (RCIC) System, and the High Pressure Coolant Injection (HPCI) System to discharge water to the reactor vessel.

2.0.7 Reactor Core Isolation Cooling System (Section 2.7)

The Reactor Core Isolation Cooling (RCIC) System supplies high pressure makeup water to the reactor vessel when the reactor is isolated from the main condenser and/or loss of the reactor feed pumps.

2.0.8 Reactor Water Cleanup System (Section 2.8)

The Reactor Water Cleanup (RWCU) System maintains reactor water quality by removing corrosion products, fission products and other impurities that end up in the reactor coolant. The RWCU System also provides a path for the removal of reactor coolant from the reactor vessel during periods of reactor startup and shutdowns.

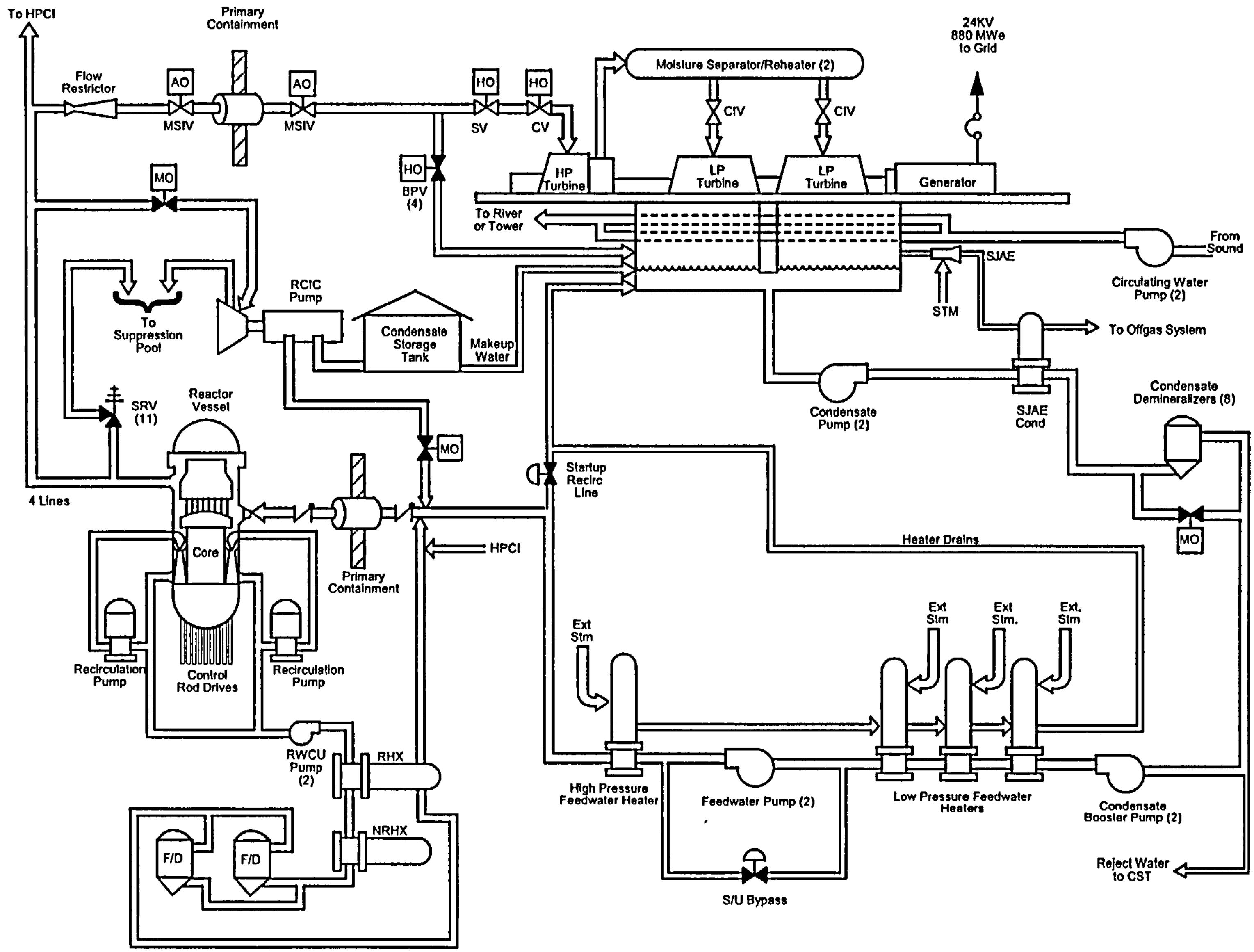


Figure 2.0-1 Simplified BWR Primary and Auxiliary Systems

0102

	Pressure (psia)	Flow (Mlb/hr)	Temperature (F)	Enthalpy (Btu/lb)
1. Core Inlet	1060	77	532	526.9
2. Core Outlet	1033	77	548	634.9
3. Separator Outlet (Steam Dome)	1020	10.5	547	1191.5
4. Steam Line (2ND Isolation Valve)	965	10.5	543	1191.5
5. Feedwater Inlet (Includes return flow)	1045	10.5	420	397.8
6. Recirc Pump Suction	1032	34.2	532	526.8
7. Recirc Pump Discharge	1206	34.2	532	527.6

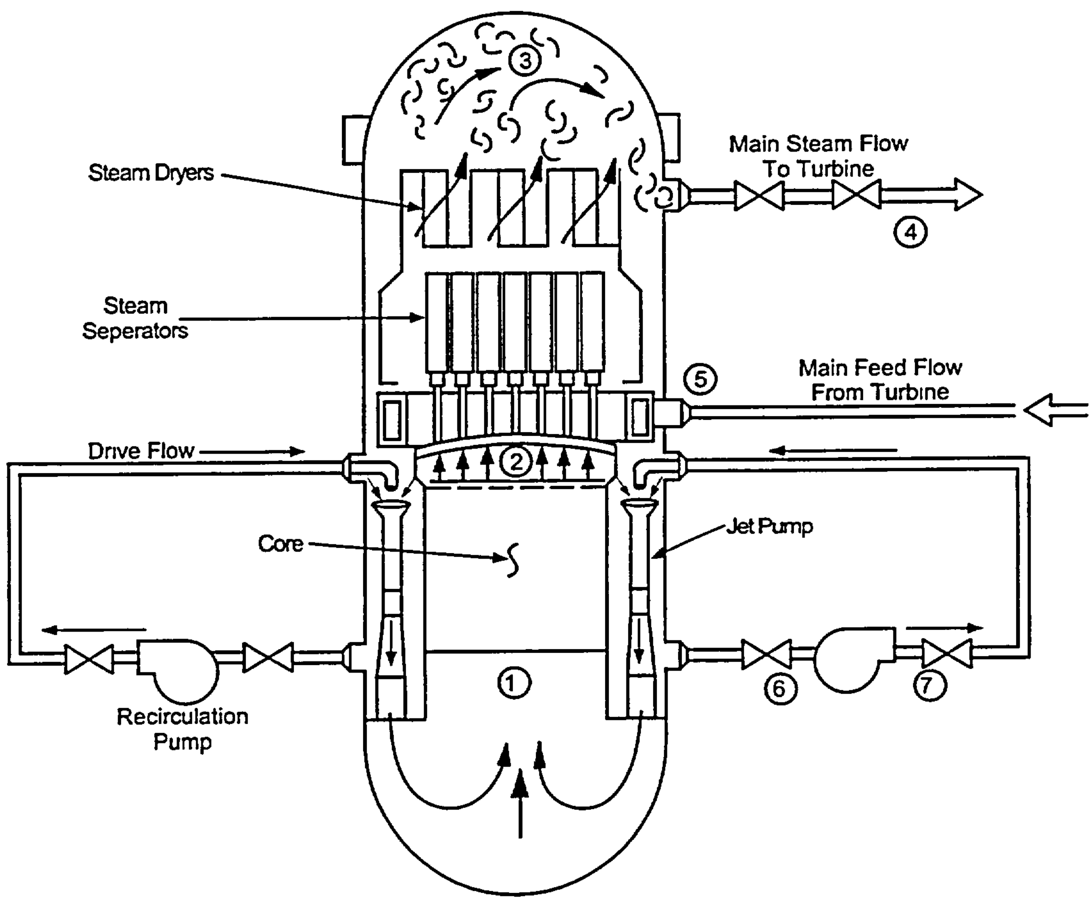


Figure 2.0-2 Operating Conditions of a BWR

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2.1 REACTOR VESSEL SYSTEM

The purposes of the Reactor Vessel System are to house and support the reactor core; to provide water circulation to the reactor core to remove generated heat; to separate the water and steam produced in the reactor core and deliver dry steam to the Main Steam System; and to provide an internal, refloodable volume to assure core cooling capability following a loss of coolant accident (LOCA).

The functional classification of the Reactor Vessel System is that of a safety related system. It also contains a component, the control rod drive housing support structure, which is an engineered safety feature.

2.1.1 System Description

The reactor vessel assembly, shown in Figure 2.1-1, consists of the reactor vessel and its internal components; including the core support structures, core shroud, moisture removal equipment, and jet pump assemblies. The various pipes that penetrate the reactor vessel and the vessel internal structure are discussed in later paragraphs of this section.

2.1.2 Component Description

The major components of the Reactor Vessel System can best be discussed when divided into several different categories:

- The reactor vessel.
- The annulus region components.
- The lower region components.
- The core region components
- The above core region components.
- The components external to the reactor vessel.

The components associated with each of these categories are discussed in the paragraphs that follow.

2.1.2.1 Reactor Vessel

The reactor vessel is mounted vertically within the drywell. The vessel of a cylindrical center section with a rounded upper and lower head. The lower head of the reactor vessel is an integral part of the main assembly with the upper head being removable to facilitate refueling operations. The vessel assembly is supported by a support skirt which is mounted to the reactor vessel support pedestal.

The reactor vessel, top and bottom head are fabricated from a low carbon steel alloy. The entire vessel assembly is designed for 1250 psig, 575°F. The inside wall of the cylindrical shell and the bottom head are clad with a stainless steel weld overlay to provide necessary corrosion resistance. The inside of the top head is not clad since it is exposed to the less corrosive steam environment.

The upper vessel head is attached to the cylindrical vessel center section (shell) by a stud and nut arrangement. To ensure a water tight seal between the head and shell flanges, two hollow concentric O-rings are mounted in matching grooves in the head flange. The O-rings are self-energized by permitting reactor pressure to act on the inside of the O-rings through slots in the O-rings. Because the O-rings are retained in the upper vessel head by clips, they can be replaced outside the refueling floor reactor cavity area. The double O-ring seal and flange design permits vessel heatup and cooldown without leakage past the second O-ring. The space between the two O-rings is tapped and piped to a pressure and level alarm system. If the inner O-ring should fail to seal, leakage is then detected by the level and/or pressure switches.

2.1.2.2 Annulus Region Components

The reactor vessel annulus region is the area inside the reactor vessel shell, outside the core shroud, and above the baffle plate. The components in this area are discussed in the paragraphs which follow.

2.1.2.2.1 Feedwater Spargers

The reactor vessel is equipped with six feedwater spargers, Figure 2.1-2, located in the mixing section above the downcomer. The feedwater spargers distribute the incoming feedwater to enhance mixing with the hot water returning from the steam separation equipment. Each sparger is fitted to a vessel nozzle penetration and manufactured to conform with the curvature of the vessel wall.

Feedwater entering the vessel through the feedwater nozzles enter through the center of the thermal sleeve. The thermal sleeve directs the feedwater to the distribution header, while preventing the relative cold feedwater from contacting the vessel wall.

The converging discharge nozzles direct feedwater from the distribution header radially upward and inward. The relatively cool feedwater in the downcomer annulus regions, subcools the water flowing to the recirculation pumps and jet pumps to guard against steam formation in the pumps.

2.1.2.2.2 Recirculation Suction Penetrations

Two recirculation suction lines penetrate the reactor vessel. The suction lines direct annulus region water to the Recirculation System (section 2.4 of this manual).

2.1.2.2.3 Recirculation Discharge Penetrations

The ten recirculation inlet penetrations route water from the recirculation pump discharge to the jet pump nozzles.

2.1.2.2.4 Jet Pump Assemblies

The jet pump assembly, shown in Figure 2.1-3, consists of one inlet riser pipe, two nozzle sections, two mixing sections, two diffuser sections, and the required restraining equipment for support. The jet pumps are arranged in two semicircular groups of ten, with each group being supplied from a separate recirculation pump.

The jet pump nozzles, suction inlets, and throat or mixing sections are joined together with a mechanical slip fit for easy removal during maintenance. The diffuser mates, via a slip fit at the top, with the mixing section and is welded to the baffle plate at the bottom. A firm force, on all the slip fit section, is provided by the hold down bolt assembly.

The design of the jet pumps and arrangement with the shroud permits reflooding the reactor core to at least the height of the jet pump suction following a loss of coolant accident (Section 2.1.3.2).

2.1.2.2.5 Baffle Plate

The baffle plate, shown in Figure 2.1-4, is welded to the reactor vessel wall and supported underneath by column members welded to the vessel bottom head. In addition to supporting the core shroud, the baffle plate provides support for the jet pump assemblies and separation between the annulus region and lower core region.

2.1.2.3 Lower Plenum Region

The lower plenum region is the area inside the reactor vessel lower head, below the baffle plate and core plate. The components in this region are discussed in the paragraphs which follow. The reactor vessel lower plenum region is illustrated in Figure 2.1-4.

2.1.2.3.1 Bottom Head Penetrations

The reactor vessel bottom head, which is welded to the vessel shell, contains numerous penetrations which consists of:

- One penetration for the bottom head drain to the Reactor Water Cleanup System.
- One penetration for pressure measurement below the core plate.
- One penetration for pressure measurement above the core plate.
- A penetration for each control rod drive mechanism; 185 total.
- A penetration for each local power range monitoring detector string; 43 total.
- A penetration for each intermediate range monitoring detector; 8 total.
- A penetration for each source range monitoring detector; 4 total.

2.1.2.3.1.1 Bottom Head Drain

The bottom head drain line penetration directs water to the Reactor Water Cleanup System (RWCU, section 2.8) to aid in the removal of suspended solids, provide a means for monitoring the water temperature in the bottom head area, and to minimize cold water stratification at low core flows.

2.1.2.3.1.2 Below Core Plate Pressure Line

The below core plate pressure sensing line is used to monitor jet pump performance and provide and input to the jet pump flow measurement network.

2.1.2.3.1.3 Above Core Plate Pressure Line

The above core plate pressure sensing line is used, along with the below core plate pressure measurement, to measure core differential pressure.

2.1.2.3.1.4 Control Rod Drive Housing

The control rod drive (CRD) housing, shown in Figure 2.1-5, are extensions of the reactor vessel bottom head. The CRD housings provide vertical and lateral support for the CRD mechanism. Each housing also transmits the weight of four fuel assemblies, a fuel support piece, and a CRD guide tube to the vessel bottom head.

2.1.2.3.1.5 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings through the core plate, Figure 2.1-6. Each tube is designed to provide lateral support for a control rod, vertical support for a four-lobed fuel support piece, and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing; which, in turn, transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head.

2.1.2.3.1.6 Incore Housings and Guide Tubes

The incore housings provide for mounting of the incore nuclear instrumentation assemblies (see Chapter 5.0 for details). The guide tubes are welded to the top of the incore housings and extend up to the core plate. The guide tubes and housings prevent jet pump flow (core flow) impingement on the nuclear instrumentation assemblies in the below core plate area, thereby eliminating possible vibration damage to these assemblies.

Each guide tube is perforated at the lower end by 4 holes to provide a cooling water path for the nuclear instrumentation assemblies. These nuclear instrumentation assemblies are loaded into the guide tubes from the top of the vessel with the assemblies extending upward from the guide tubes into the bottom of the top guide.

2.1.2.4 Core Shroud Region Components

The core shroud region is the area bounded at the bottom by the core plate, at the top by the shroud head, and circumferentially by the core shroud. The components of this region are discussed in the paragraphs that follow.

2.1.2.4.1 Core Shroud

The core shroud, shown in Figure 2.1-5, is a two inch thick, cylindrical stainless steel assembly which surrounds the core. The shroud provides the following:

- A floodable volume following a loss of coolant accident (LOCA).
- A barrier to separate or divide the upward core flow from the downward flow in the downcomer or annulus region.
- A vertical and lateral support for the core plate, top guide and shroud head.
- A mounting surface for the core spray spargers (CS, Section 10.3).
- Aids in forming the core discharge plenum.

2.1.2.4.2 Core Plate

The core plate, shown in Figure 2.1-5, consists of a circular, horizontal stainless steel plate with vertical stiffener plate members below the horizontal plate. The core plate acts as a partition to force the majority of the core flow into the control rod guide tubes where it will be directed to the fuel assemblies. The core plate also provides vertical and lateral support for 24 peripheral fuel assemblies via their fuel support piece.

2.1.2.4.3 Top Guide

The top guide is a lattice assembly formed by a series of stainless steel plates joined at right angles to form square openings for control cells. Along the periphery are smaller openings which accommodate the peripheral fuel assemblies. Cutouts are provided on the bottom edge of the top guide, at the junction of the cross plates, to accommodate the spring loaded upper ends of the neutron instrument assemblies and source holder. Support for the top guide is provided by a rim near the top end of the shroud and is held in place by a nut and bolting arrangement.

2.1.2.4.4 Fuel Support Pieces

The fuel support pieces, Figure 2.1-7, are of two basic types, peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core plate, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains a replaceable orifice assembly designed to assure proper coolant flow to the fuel assembly. The four-lobed fuel support pieces support four fuel assemblies each and are provided with orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed fuel support pieces rest in the top of the control rod guide tubes and are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed fuel support pieces. A control rod and the four fuel assemblies which immediately surround it represent a control cell.

2.1.2.4.5 Core Spray Spargers

Two core spray pipes penetrate the reactor vessel 180° apart. Once inside the vessel, the pipes divide and are routed 90° in each direction from the point of vessel entry. The pipes (Figure 2.1-8) are directed down and then inward to penetrate the upper core shroud just below the shroud head flange. After penetrating the shroud, the lines are again divided and proceed around the inside of the upper shroud in two semicircular headers. Spray nozzles connected to these headers are adjusted to provide the correct spray distribution to the fuel assemblies.

2.1.2.4.6 Shroud Head

The shroud head, Figure 2.1-9, is rounded in shape and an integral part of the steam separator assembly. The top of the shroud head contains an array of penetrations used to channel the steam/water mixture exiting the core to the steam separator standpipes which are welded to the penetrations. The shroud head is attached to the shroud, with a special bolting arrangement, to form the core discharge plenum.

2.1.2.5 Moisture Separation Assemblies

Moisture or steam separation is accomplished internal to the reactor vessel via a steam separator and steam dryer. The steam separating components are discussed in the paragraphs which follow.

2.1.2.5.1 Steam Separator Assembly

The steam separator assembly, Figure 2.1-9, consists of individual stainless steel axial flow cyclone separators welded to the top of standpipes. Within each separator, the steam water mixture passes swirl vanes which impart a spin to establish a vortex separating the water from the steam. The steam exits the top of the separator with a steam quality between 90 to 95%. The separated water exits from under the separator cap and flows out between the standpipes, draining into the downcomer annulus region, joining the feedwater.

2.1.2.5.2 Steam Dryer Assembly

The steam dryer assembly, Figure 2.1-10, dries the wet steam leaving the steam separators to greater than 99.9% quality. The dryer assembly also provides a seal between the wet steam area and the dry steam flowing to the steam lines. The seal is formed by the steam dryer assembly seal skirt extending below the normal water level.

The individual dryer sections force the steam to be directed horizontally through the dryer panels. The steam is forced to make a series of rapid changes in direction while traversing the panels. During these direction changes, the heavier drops of entrained moisture are forced to the outer walls where moisture collection hooks catch and drain the liquid to collection troughs. From the collection troughs the liquid is directed to the annulus area via drain tubes.

2.1.2.6 Main Steam Outlet Penetrations

Four, twenty-four inch diameter steam line nozzles are installed to direct the steam out of the reactor vessel.

2.1.2.7 Vessel Head Penetrations

The reactor vessel head contains three penetrations, shown in Figure 2.1-1, of which only two are normally used after completion of the initial startup testing program. The first penetration closest to center and commonly called the head vent line, provides the dual function of venting noncondensible gases from the reactor vessel head area and providing a sensing line for shutdown level indication. During operation at temperatures less than the boiling temperature, the noncondensible gases are vented to the drywell equipment drain sump through two motor operated valves. The second penetration, which is approximately 30 inches radially off center, is used by the Residual Heat Removal System (RHR, Section 10.4) and termed the head spray line.

2.1.2.8 Vessel External Components

Components external to the reactor vessel are discussed in the paragraphs which follow.

2.1.2.8.1 Reactor Vessel Support Skirt

The support skirt, shown in Figure 2.1-1, is welded at the top to the reactor vessel bottom head. The support skirt is anchored at the bottom to the vessel support pedestal via a ring girder and anchoring bolts. The support skirt provides vertical support for the reactor vessel, its internal components, the fuel and control rods, and the moderator in the vessel.

2.1.2.8.2 Reactor Vessel Pedestal

The concrete and steel reactor pedestal, shown in Figure 2.1-1, is constructed as an integral part of the reactor building foundation. Steel anchor bolts, set in the concrete, extend through a bearing plate and secure the flange of the reactor vessel support skirt to the bearing plate and thus to the support pedestal. The reactor pedestal also supports the biological shield.

2.1.2.8.3 Biological Shield

The biological shield (Figure 2.1-11) is a cylindrical structure of high density concrete between interior and exterior steel liners. The base of the shield wall attaches to the reactor pedestal, while the top of the shield wall is free. The biological shield surrounds the reactor vessel to provide shielding for personnel and equipment during normal power operation and shutdown conditions. In addition, the shield wall provides structural support for vessel insulation, access platforms in the area of the reactor vessel, and other equipment supports. Openings are provided in the shield to permit passage of required vessel piping.

2.1.2.8.4 Reactor Vessel Insulation

Heat losses to the drywell atmosphere are minimized by insulation panels provided for the reactor vessel shell, top and bottom heads, and vessel nozzles. The insulation panels for the cylindrical vessel shell sections are held in place by insulation supports attached to the biological shield. Sections of the insulation are removable to permit access for vessel inspection

2.1.2.8.5 CRD Housing Support Structure

Located below the reactor vessel is a control rod drive housing support structure, shown in Figure 2.1-1, which prevents the ejection of a control rod in the unlikely event of a control rod drive housing failure with the reactor pressurized. This network consists of support beams on the inside of the reactor pedestal with hanger rods and spring washers suspended from the beams. Grid clamps, grid plates, and support bars are bolted to the hanger rack to vertically support the bottom end of each housing and drive.

2.1.3 System Features and Interrelations

A short discussion of the system features and interrelations this system has with other plant systems is given in the paragraphs that follow.

2.1.3.1 Normal Operation (Figure 2.1-12)

During normal operation, the recirculation pumps and jet pumps provide forced flow of coolant through the core. Approximately 90% of this flow enters the fuel assemblies while a designed 10% bypasses the fuel to cool other incore components. The water absorbs the heat energy of the fuel which increases the water temperature to saturation and boils a portion of the total core flow.

The steam and water mixture exiting the core is forced to pass through the two stages of moisture removal in the steam separator and dryers. The water removed in moisture separation flows back to the annulus area where it mixes with the incoming feedwater. The dry steam exits the reactor vessel through the four main steam lines. Incoming feedwater enters the reactor vessel at six different penetrations and is distributed via feedwater spargers.

2.1.3.2 Core Floodability (Figure 2.1-13)

The Emergency Core Cooling Systems (ECCS) and the reactor vessel design are compatible to ensure that core can be adequately cooled following a loss of coolant accident (LOCA).

The worse case loss of coolant accident, with respect to core cooling, is a recirculation line break with the reactor at full power. In this case, the reactor vessel water level rapidly decreases uncovering the core. However, several systems automatically provide makeup water to the reactor core within the shroud. Water level increases until it reaches the level of the top of the jet pump mixing sections. Water then spills out of the jet pump suction into the annulus area and out through the broken recirculation line. The jet pump suction elevation is approximately at 2/3 of the height of the core.

If flooding of the reactor vessel is accomplished within a specified time frame, and if level is maintained at the 2/3 core coverage point, the core will be adequately cooled and the integrity of the fuel cladding will be maintained. The lower 2/3 of the core will be cooled because it is flooded with water and the upper 1/3 of the core will be cooled by a mixture of steam and water flowing upward because of the vigorous boiling in the lower 2/3 of the core.

2.1.3.3 System Interrelations

A short discussion of the interrelations this system has with other plant system is given in the paragraphs that follow.

2.1.3.3.1 Fuel and Control Rods System(Section 2.2)

The fuel and control rods are located in the core region and are vertically supported by reactor vessel components.

2.1.3.3.2 Control Rod Drive System (Section 2.3)

The Control Rod Drive (CRD) System provides cooling and hydraulic driving water to the control rod drive mechanisms mounted in the 185 CRD housings of the reactor vessel.

2.1.3.3.3 Recirculation System (Section 2.4)

The Recirculation System provides forced circulation of the reactor coolant to yield higher reactor power than would be possible under natural circulation conditions. The external pumping loop of the Recirculation System takes suction from the vessel annulus and discharges to the jet pump risers.

2.1.3.3.4 Main Steam System (Section 2.5)

The four main steam lines provide a steam path from the reactor vessel to the main turbine and other balance of plant steam loads. Overpressure protection is also provided for the reactor vessel by the 13 safety/relief valves mounted on the steam lines. The reactor head area is also vented to the Main Steam System.

2.1.3.3.5 Condensate and Feedwater System (Section 2.6)

The Condensate and Feedwater System provides high purity water to the reactor vessel to replace the steam generated and sent to the turbine.

2.1.3.3.6 Reactor Water Cleanup System (Section 2.8)

The Reactor Water Cleanup System maintains the reactor coolant at high purity and provides a means of draining water from the reactor vessel. Suction is taken from the recirculation loops and bottom head drain, and return is through the feedwater lines.

2.1.3.3.7 Reactor Vessel Instrumentation System (Section 3.1)

The reactor vessel level, pressure, and flow instrumentation uses instrument lines which penetrate the reactor vessel. Reactor vessel temperature instrumentation uses thermocouple pads at various vessel locations and the bottom head drain.

2.1.3.3.8 Neutron Monitoring System (Section 5.0)

The Neutron Monitoring System detectors penetrate the reactor vessel, housings. The detectors are located inside the reactor core.

2.1.3.3.9 Core Spray (CS) System (Section 10.3)

The Core Spray System provides for low pressure spraying of the core in the event of a LOCA. The water enters the reactor vessel via spray spargers located above the top guide.

2.1.3.3.12 Residual Heat Removal System (Section 10.4)

The low pressure coolant injection (LPCI) mode provides flooding water to the reactor vessel via the recirculation line.

2.1.4 Summary

Classification - Safety related system

Purposes

- To house and support the reactor.
- To provide water circulation to the reactor core to remove generated heat.
- To separate the water and steam mixture produced in the reactor core and deliver dry steam to the main steam system.
- To provide an internal, refloodable volume to assure core cooling capability following a loss of coolant accident (LOCA).

Components

Reactor Vessel; Feedwater spargers; surveillance sample holder; jet pump assemblies; core plate; top guide; CRD housing; fuel support piece; guide tubes; moisture separator and drywell; biological shield; shroud; baffle plate; various penetrations.

System Interrelations

Fuel and Control Rods System; Recirculation System; Condensate and Feedwater System; Main Steam System; Reactor Water Cleanup System; Standby Liquid Control System, High Pressure Coolant Injection System,Core Spray System; Neutron Monitoring System; Control Rod Drive System.

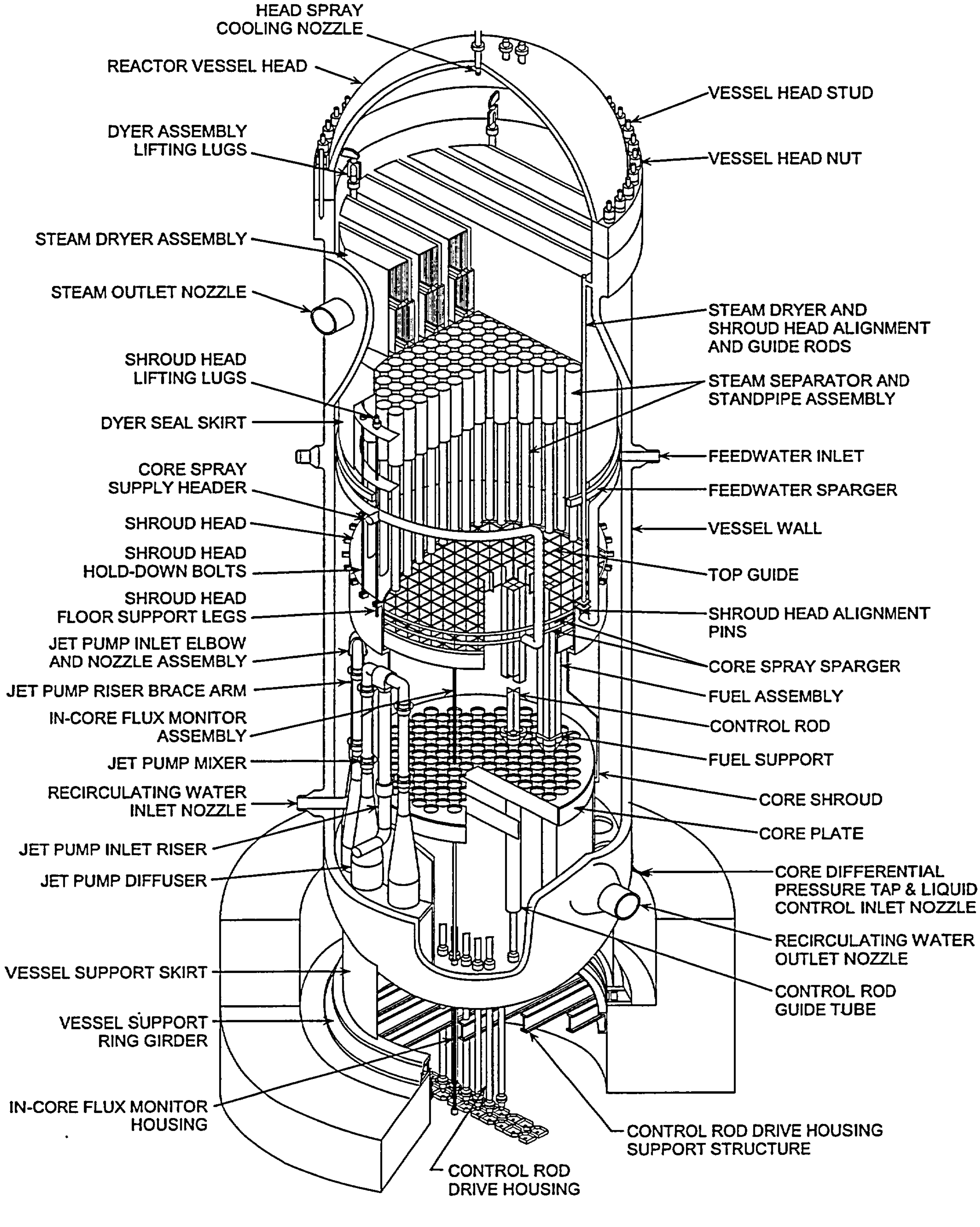

Figure 2.1-1 Reactor Vessel Cutaway

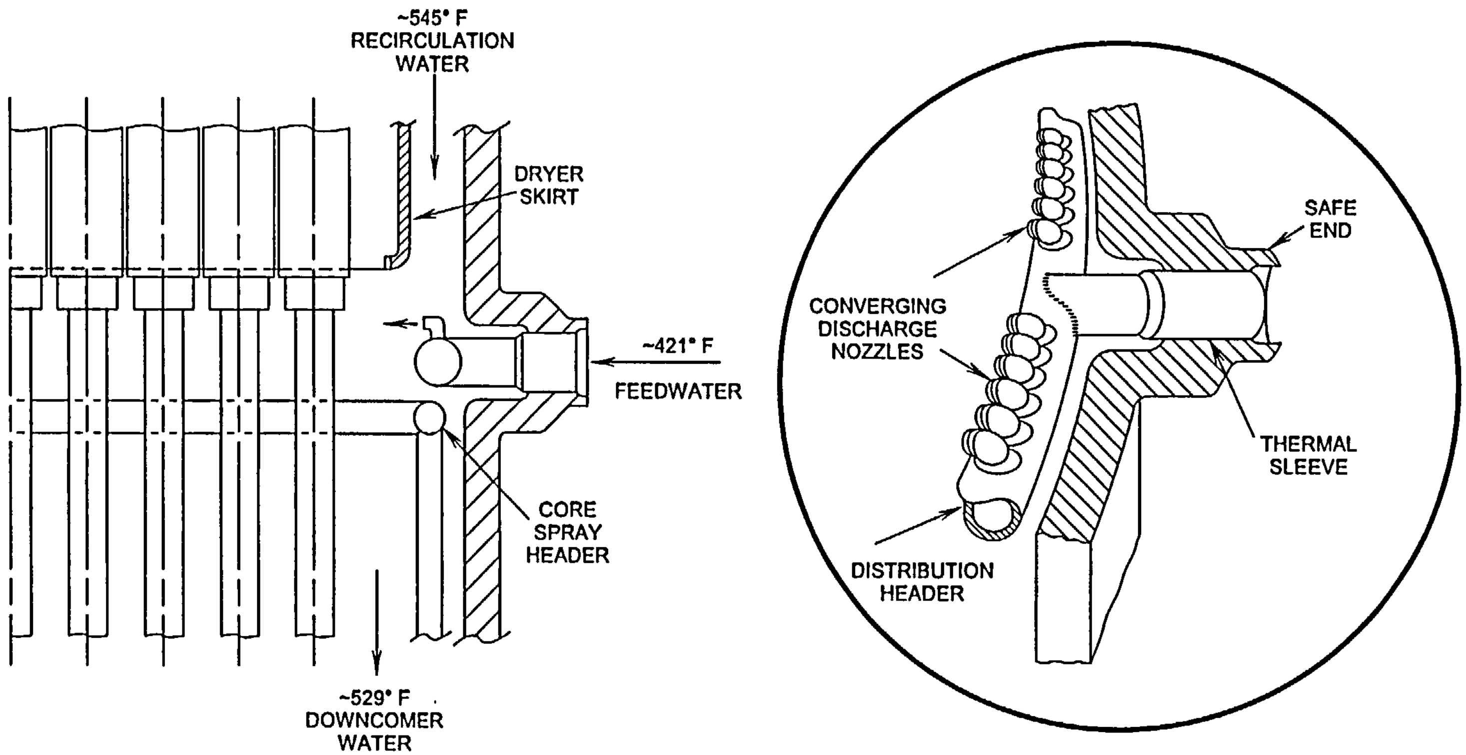

Figure 2.1-2 Typical Feedwater Sparger

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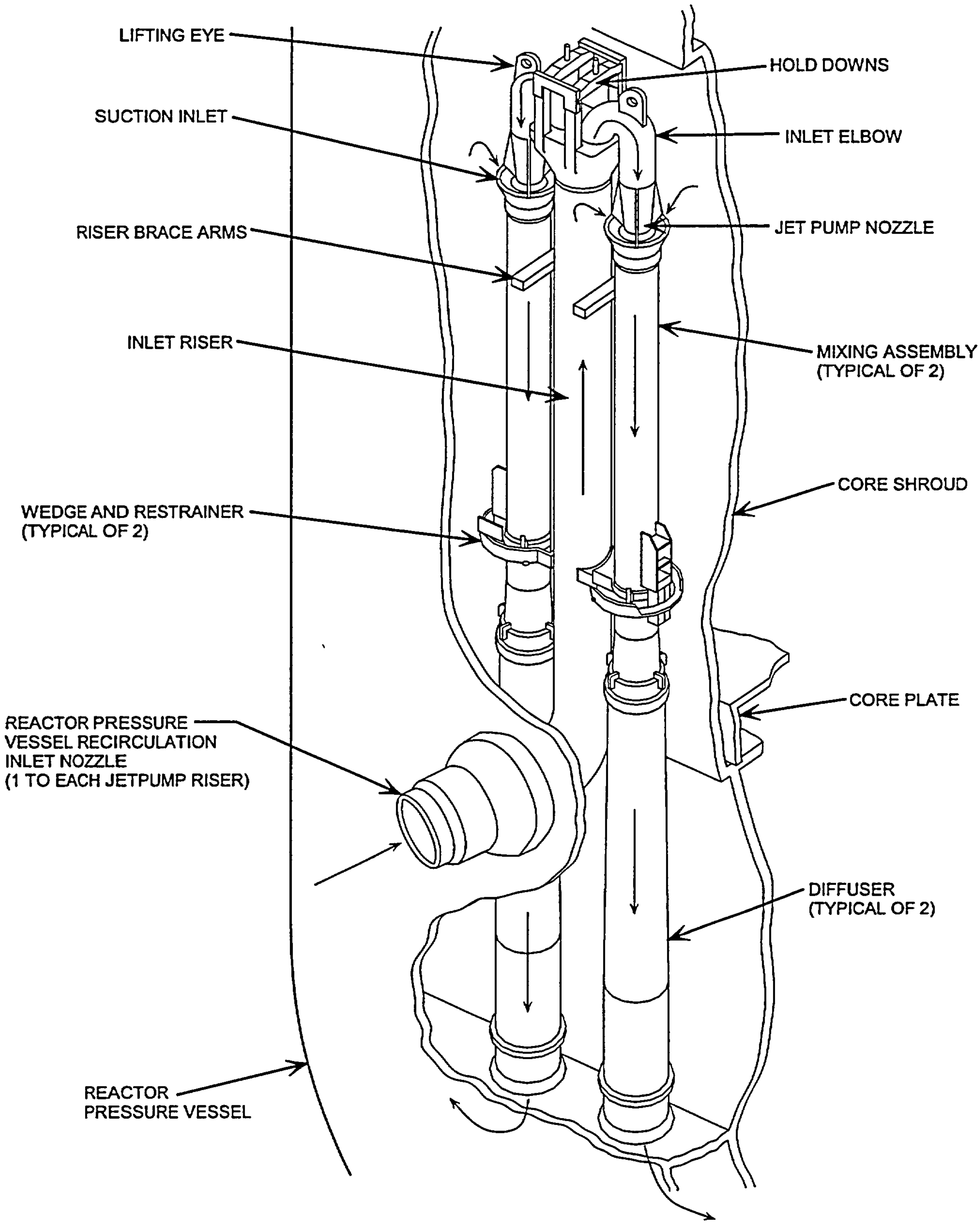


Figure 2.1-3 Jet Pump Assembly

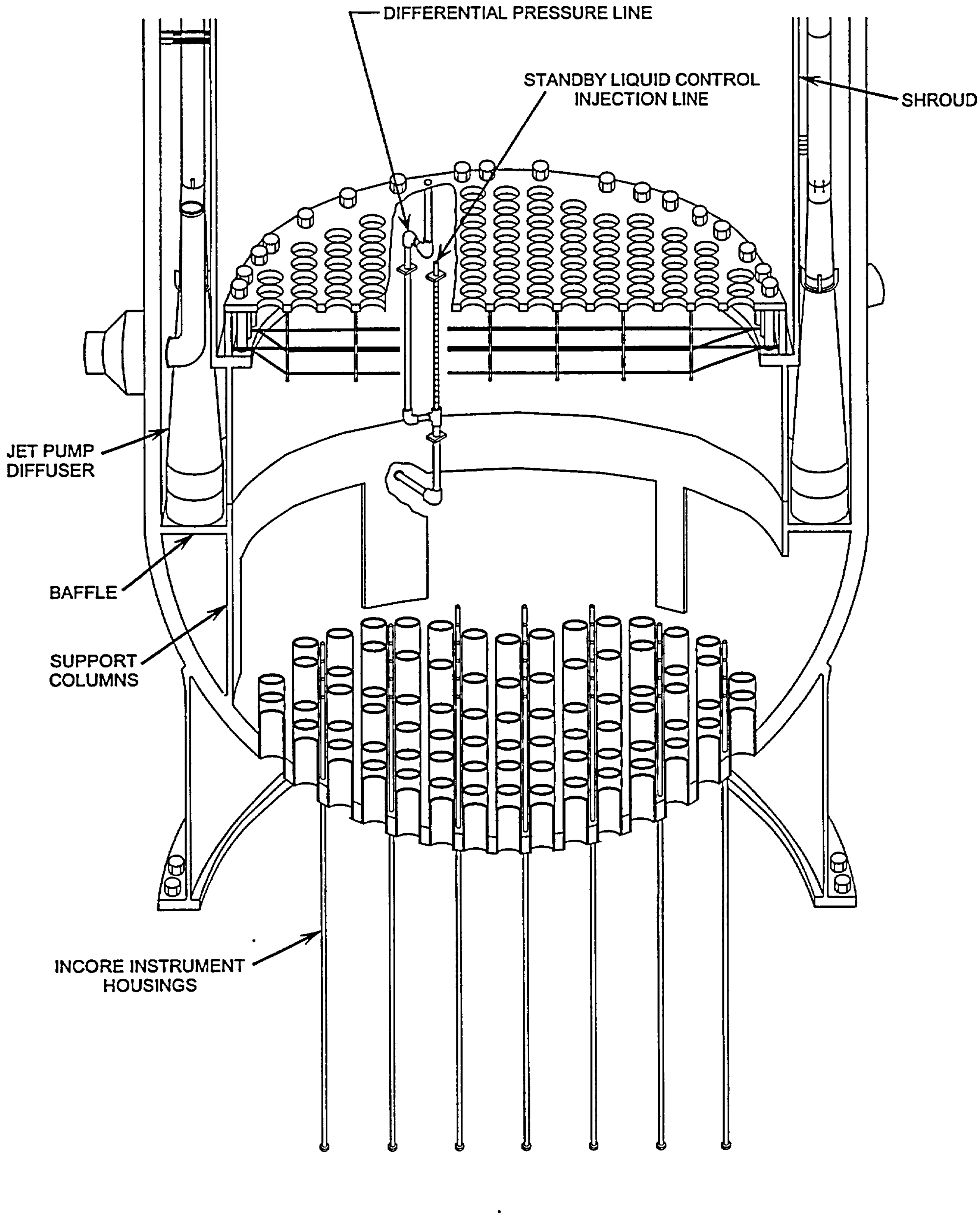


Figure 2.1-4 Differential Pressure, Standby Liquid Control Line, and Shroud Support

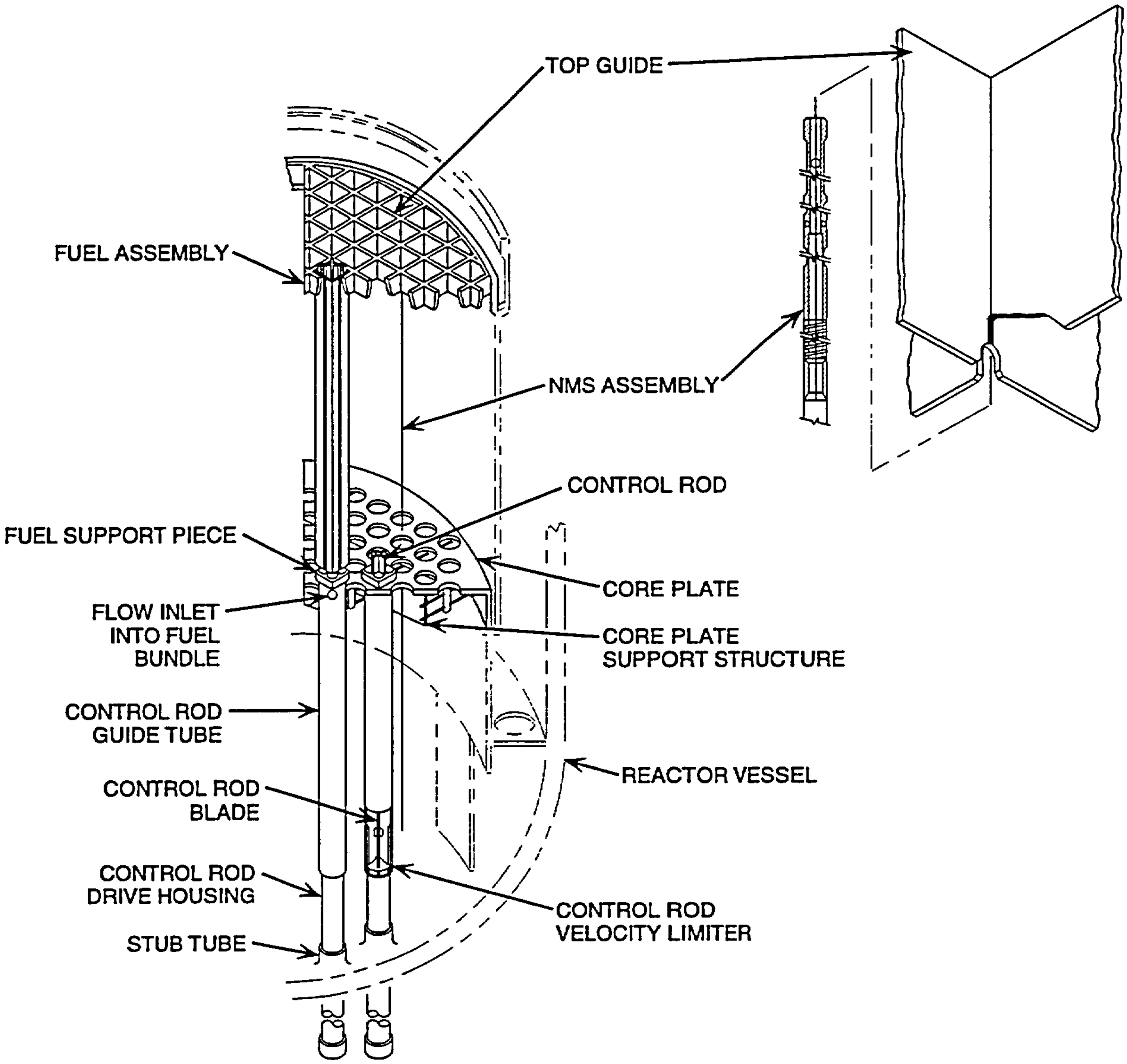


Figure 2.1-5 Shroud Intervals

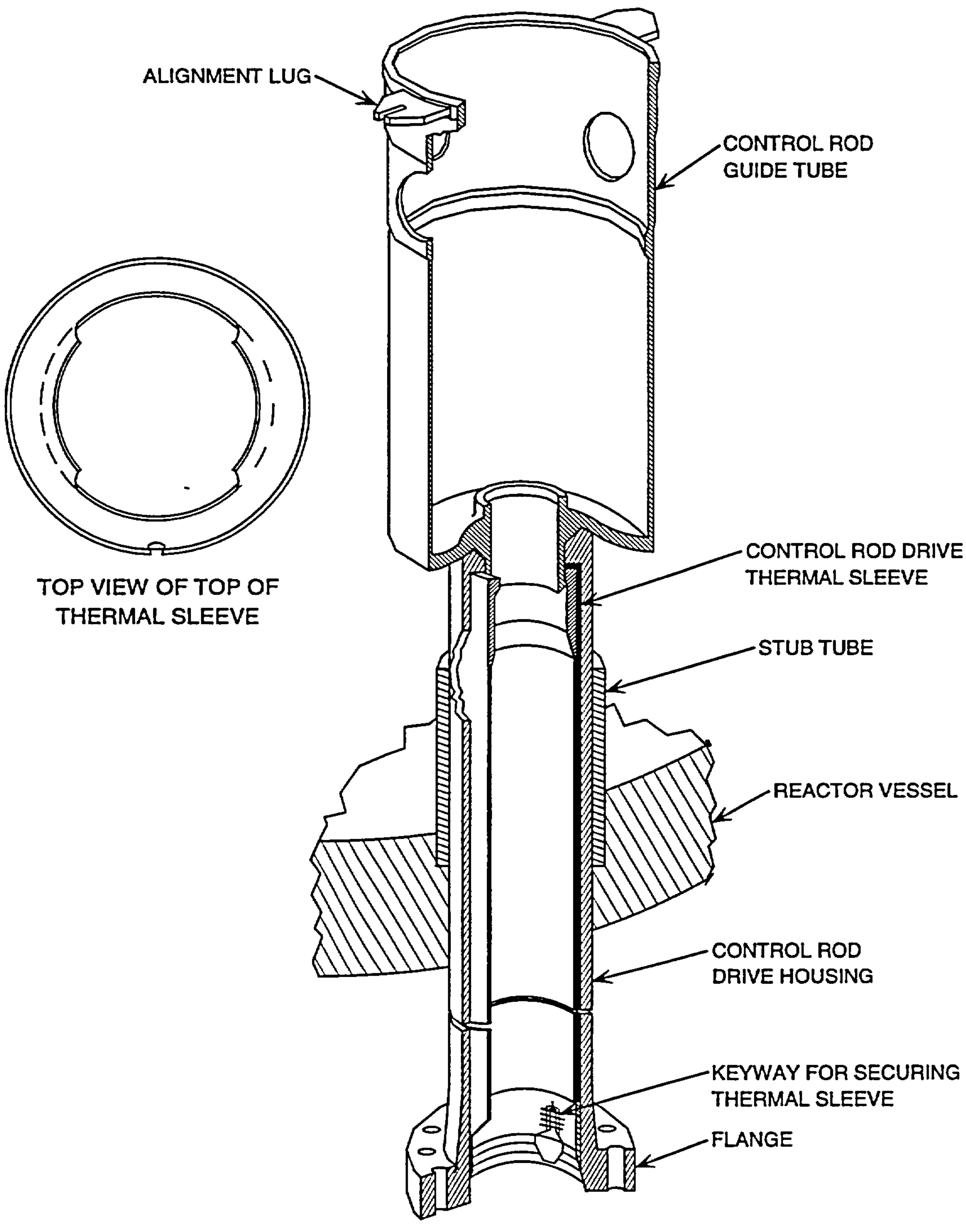


Figure 2.1-6 Control Rod Guide Tube To Control Rod Drive Housing Attachment

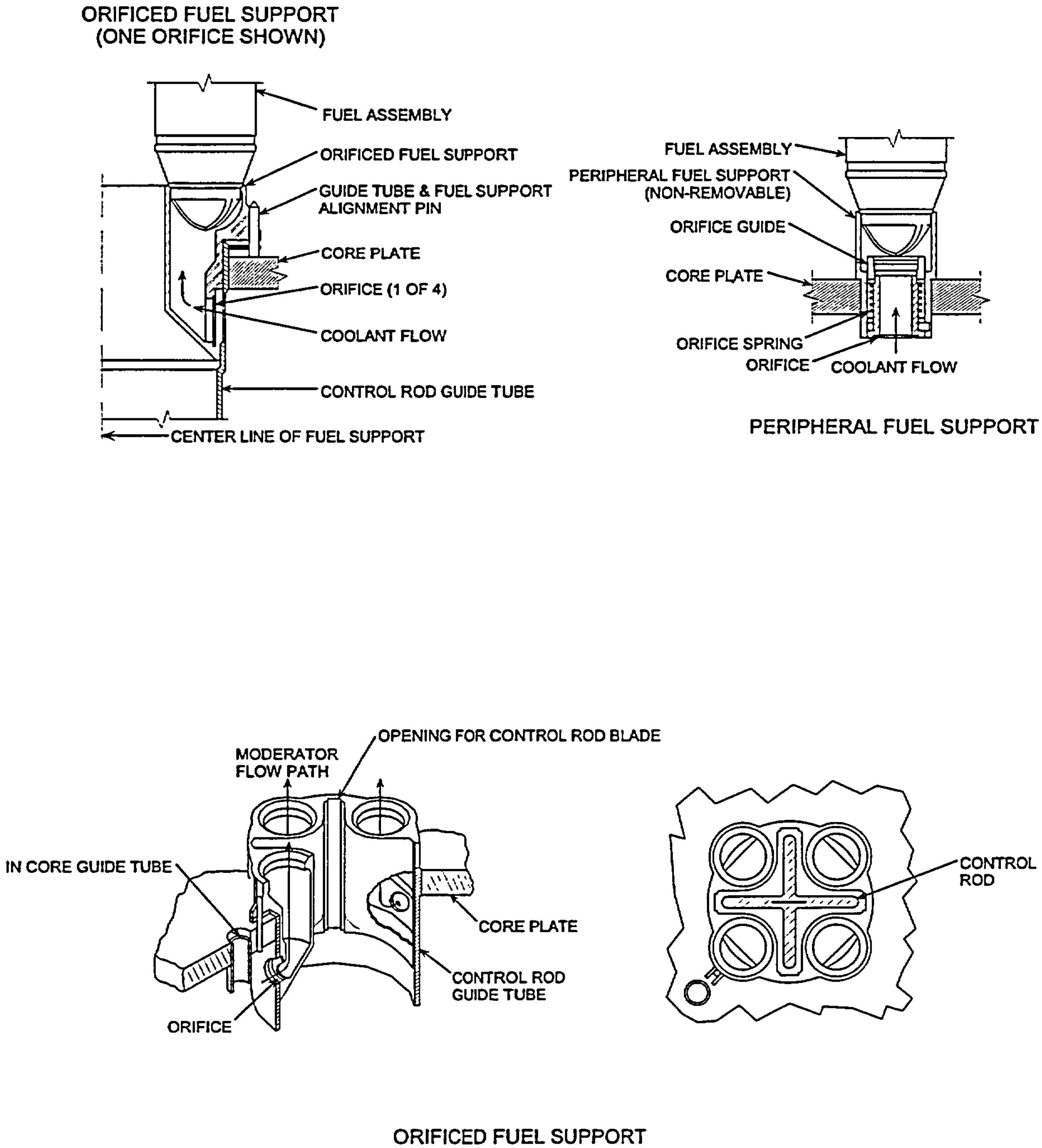
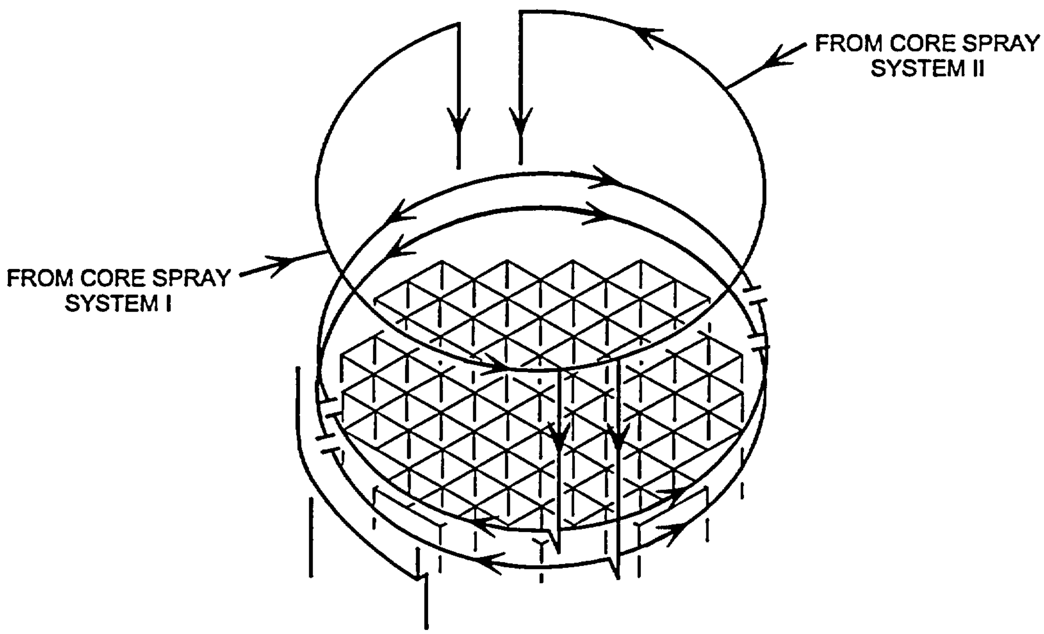


Figure 2.1-7 Fuel Support Pieces



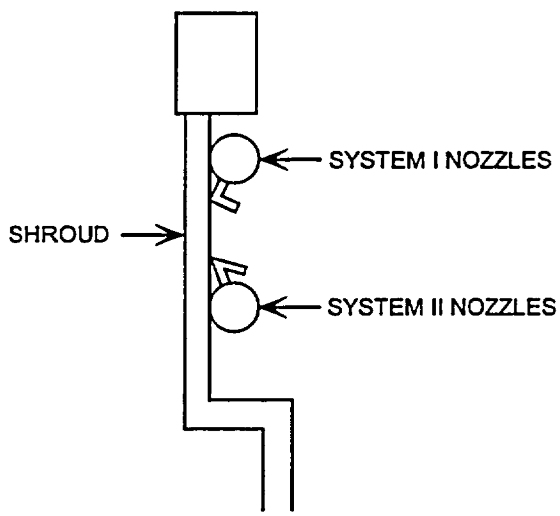


Figure 2.1-8 Core Spray Piping In Vessel

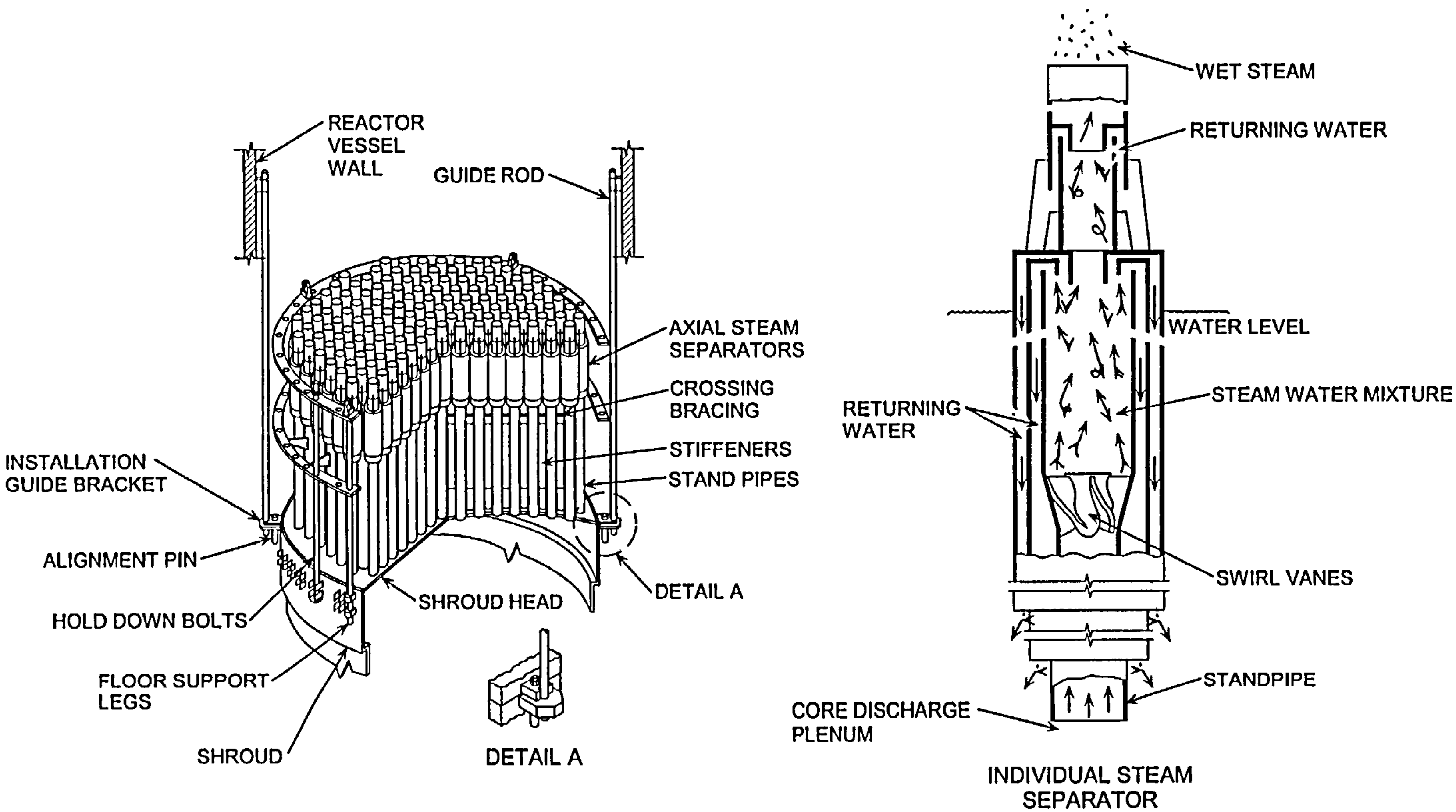


Figure 2.1-9 Shroud Head and Steam Separator Assembly

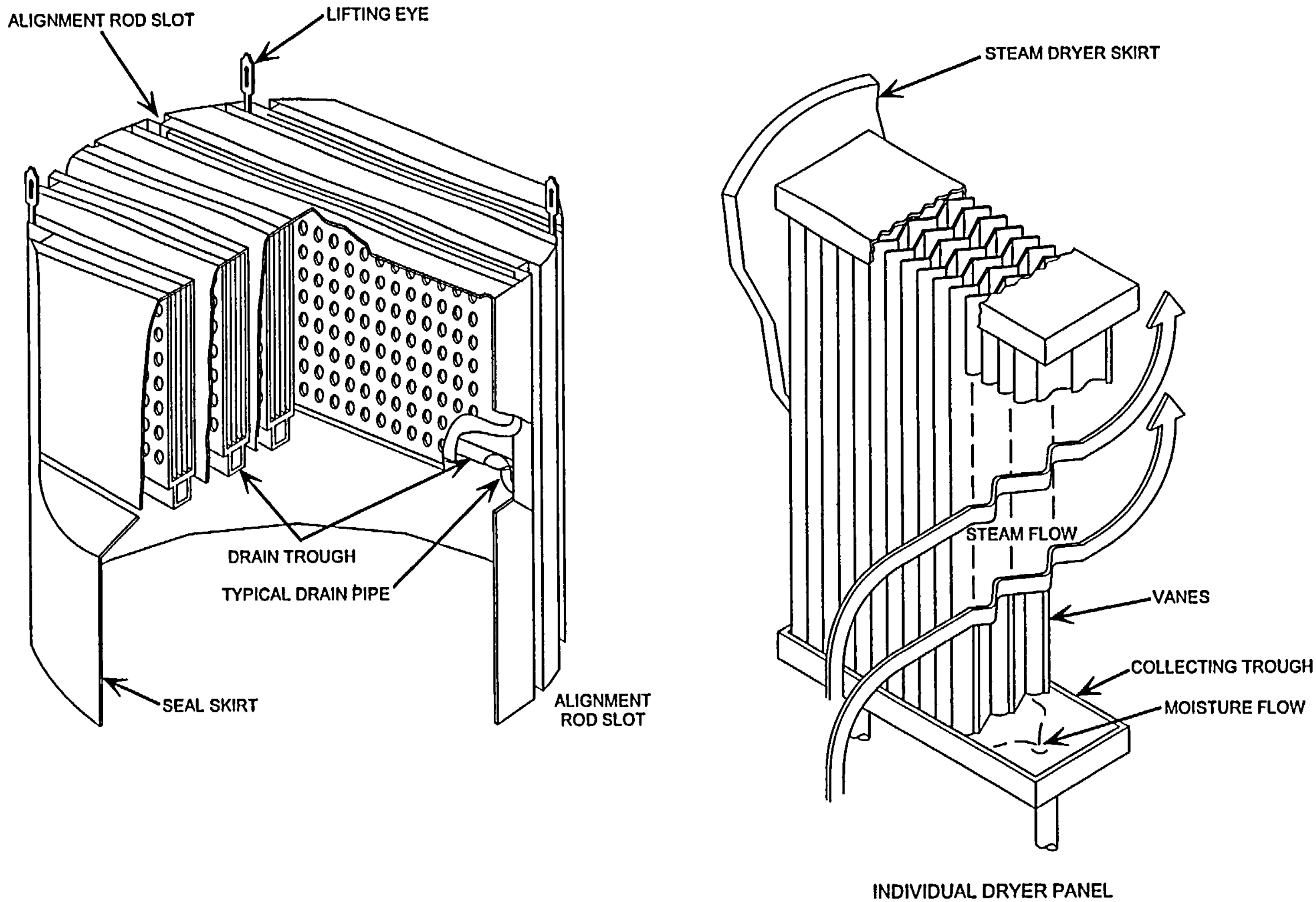


Figure 2.1-10 Steam Dryer Assembly

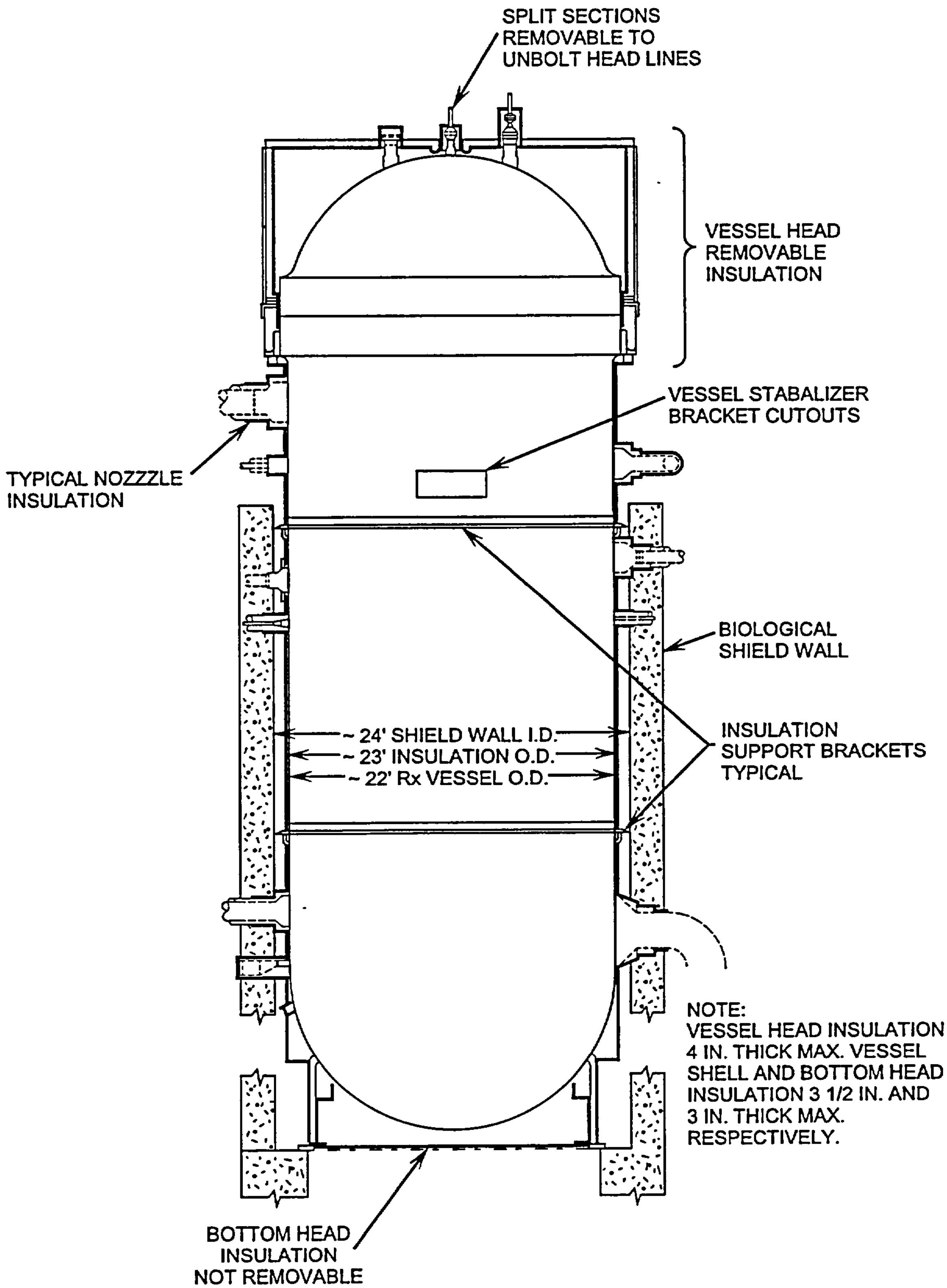


Figure 2.1-11 Reactor Vessel Insulation

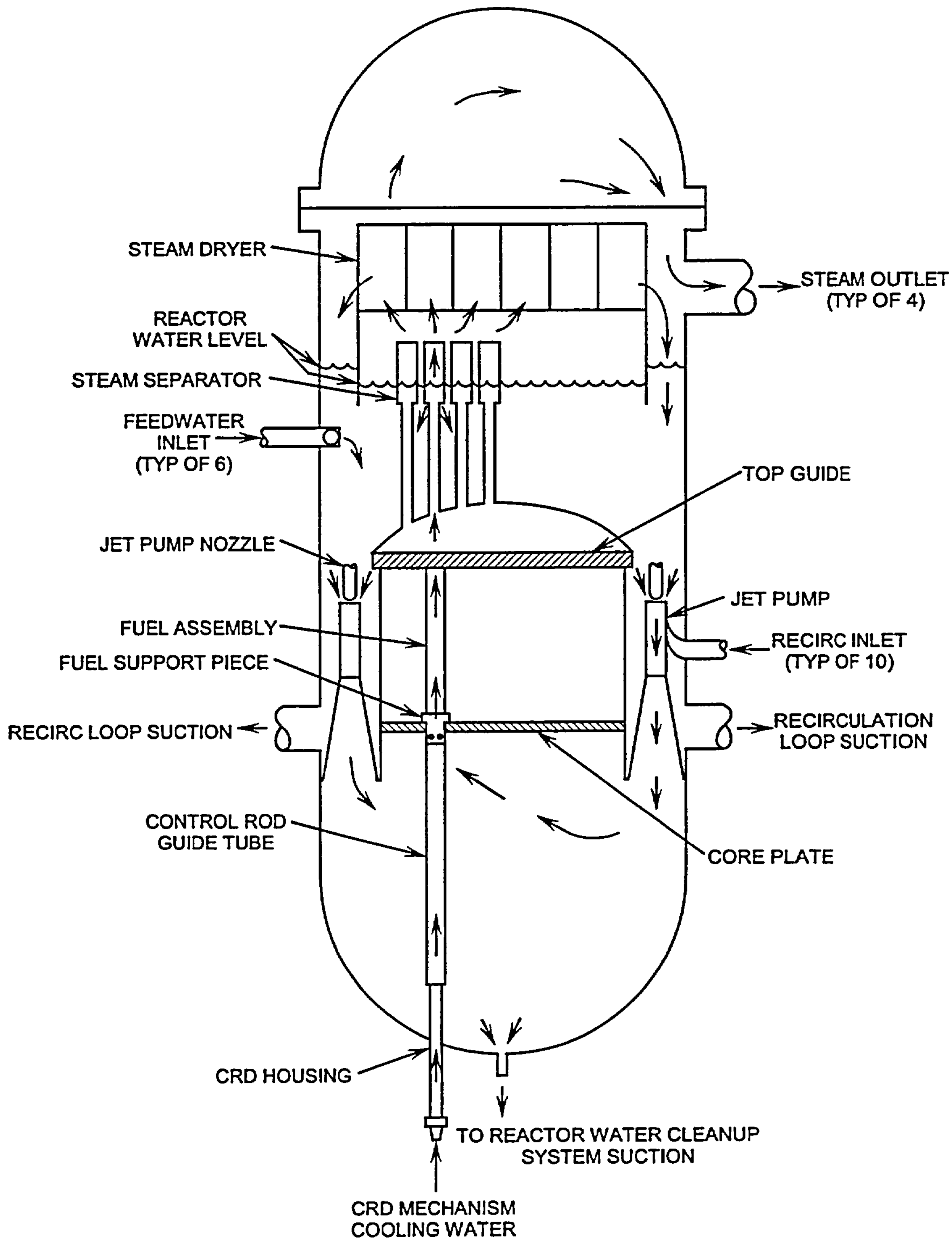


Figure 2.1-12 Reactor Vessel Flow Paths

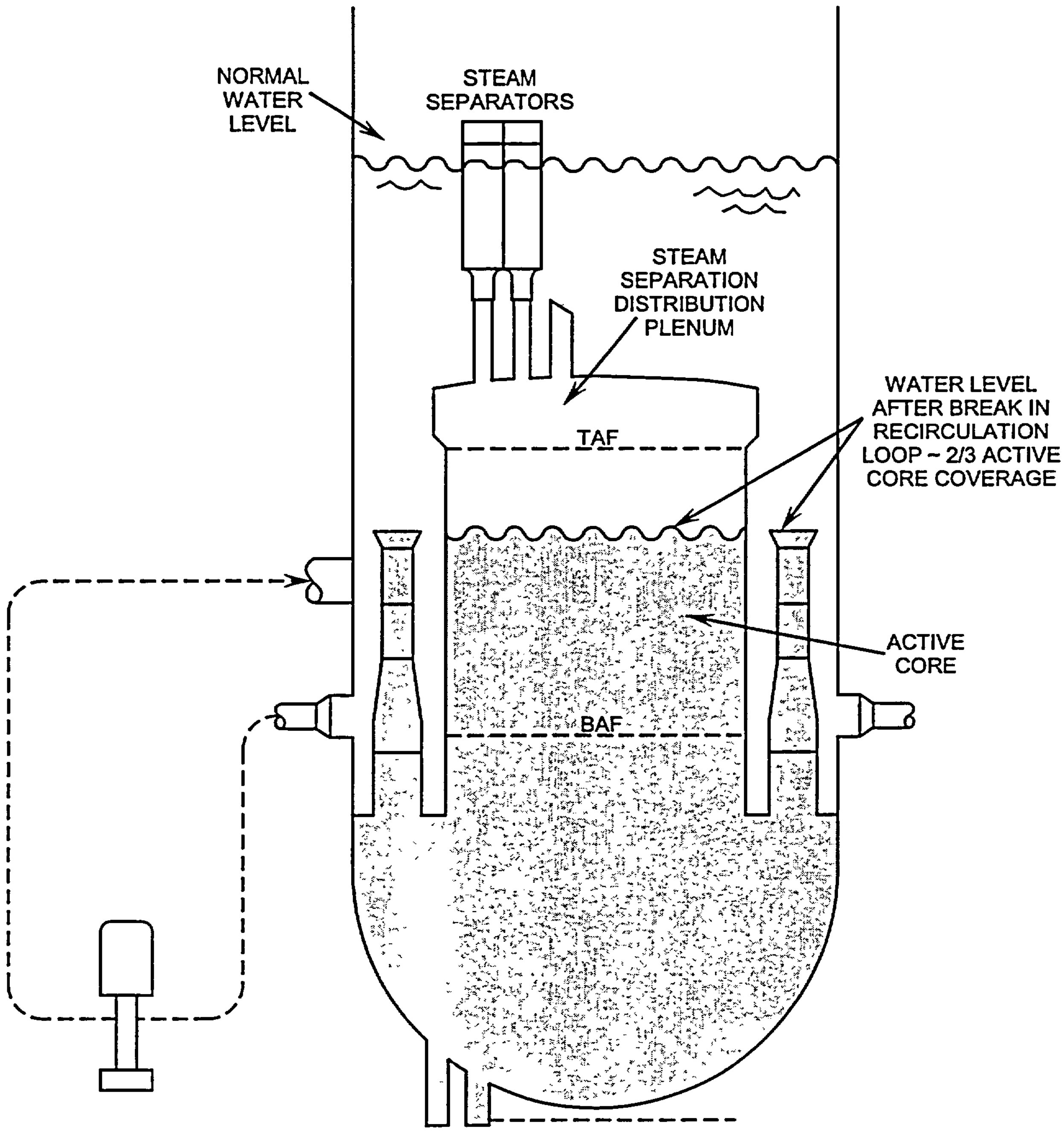


Figure 2.1-13 Core Flooding Capability of Recirculation System

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2.2 FUEL AND CONTROL RODS SYSTEM

The purpose of the reactor fuel is to generate energy from a nuclear fission reaction to provide heat for steam generation.The purposes of the control rods are:

1. To control reactor power level
2. To control reactor power distribution
3. To provide emergency shutdown capability.

The functional classification of the Fuel and Control Rods Systems is that of a safety related system. It also contains a component, the control rod velocity limiter, which is an engineered safety feature.

2.2.1 System Description

The reactor core is arranged into fuel cells. Each fuel cell consists of a control rod and the four fuel assemblies which immediately surround it. The four fuel assemblies are supported by a fuel support piece. Around the outer edge of the core, certain fuel assemblies not immediately adjacent to a control rod are supported by individual peripheral fuel support pieces.

The reactivity of the core is regulated by movement of bottom entry control rods. A control rod blade consists of a sheathed, cruciform array of vertical neutron absorber rods. The control rods are inserted and withdrawn from the bottom of the core by the control rod drive mechanisms (Section 2.3).

To limit the rate of free fall of a control rod from its fully inserted position to full out, each control rod is equipped with a velocity limiter located below the active poison section.

2.2.2 Component Description

The major components of the Fuel and Control Rods are discussed in the paragraphs which follow.

2.2.2.1 Fuel Assembly (Figure 2.2-1)

A fuel assembly consists of a fuel bundle and the fuel channel which surrounds it. Each fuel assembly has a dry weight of approximately 715 lbs. and an overall length of 176 inches. The assemblies are arranged in the reactor core to approximate a right circular cylinder.

2.2.2.1.1 Fuel Bundle

A fuel bundle contains fuel rods and one or more water rods which are spaced and supported in a square array by fuel rod spacers and upper and lower tie plates. The square array arrangement is dependent on the fuel manufacturer and the frequently changing design. BWR plants utilize three major fuels manufacturers, (General Electric Company, Advanced Nuclear Fuels Corporation, and Brown Boveri). With the constant advancements in fuel designs it would be very difficult to maintain an up to date description of all fuel types and designs. Therefore, this chapter will discuss only the fuel bundle most widely used by BWRs.

2.2.2.1.2 Fuel Channel

The fuel channel provides a barrier to separate bundle flow from bypass flow, a bearing surface and guide for the control rod blades, and protects the fuel rods during fuel handling operation.

The fuel channels are fabricated from zircaloy-4 because of its good corrosion resistance. The bottom end of the fuel channel makes a sliding seal fit with the lower tie plate by means of finger springs. At the top of the channel, two diagonally opposite corners have welded tabs which support the weight of the channel from raised posts on the upper tie plate. One of the raised posts has a threaded hole to attach the channel in place with a channel fastener assembly.

Channel to channel spacing to provide a passage for the control rod blade is maintained with the use of spacer buttons and the channel fastener assembly. The spacer buttons are located at the top of the fuel channels, facing each other across the control rod passage area.

2.2.2.1.3 Fuel Rods (Figure 2.2-2)

There are three different types of rods used in a 8x8 fuel bundle: 54 standard fuel rods, eight fueled tie rods and two water rods.

The standard fuel rods are approximately 160 inches in length with an active fuel length of 150 inches. The active fuel length consists of high density ceramic fuel pellets enriched in uranium-235 with some pellets having a uranium-gadolinium mixture. The remaining ten inches of the fuel rod contains a free volume, plenum spring and a hydrogen getter. The free volume, called the plenum volume, is designed to accommodate gaseous fission products which are released from the fuel pellets over the design life of the fuel bundle. The plenum spring maintains a constant compressive force on the fuel pellets to maintain axial contact between the pellets. The hydrogen getter provides protection against moisture or hydrogenous impurities in the fuel rod. Both ends of a standard fuel rod are sealed by end plugs which slip fit into the upper and lower tie plates.

The fueled tie rods differ from the standard fuel rods in that the lower and upper end plugs are threaded. The lower end plugs thread into the lower tie plate while the upper end plugs extend through the upper tie plate and are held in place by locking nuts. The third and sixth rods along each side of the fuel bundle are fueled tie rods. The fueled tie rods hold the fuel bundle together and support the weight of the fuel bundle during fuel handling operations

The purpose of the water rods is to maintain orientation of the seven fuel spacers and provide for better neutron economy. The two water rods are hollow zircaloy-2 tubes equipped with squarebottom end plugs to prevent rotation and assure proper location of the water rods within the fuel bundle. Several holes are drilled through the tube wall at the top and bottom of the water rods to allow coolant to flow freely throughout the rod. One of the two water rods contains welded tabs along its outer surface to lock the fuel bundle spacers in the required axial position.

2.2.2.1.4 Lower Tie Plate

The lower tie plate, manufactured from a stainless steel casting, positions the fuel bundle rods laterally and transfers the weight of the assembly to the fuel support piece. The nose piece is used to guide the bundle into place during refueling operations and to direct coolant flow up through the fuel assembly.

A hole drilled into two sides of the lower tie plate provide the bundle bypass flow used to cool the incore nuclear instruments.

2.2.2.1.5 Upper Tie Plate

The upper tie plate, also manufactured from a stainless steel casting, provides alignment and support for the top of the rods within a bundle, alignment and a mating surface for the fuel channel, and a means to attach the fuel channel to the bundle.

The lifting handle, which is an integral part of the upper tie plate, is used for fuel assembly handling during refueling operations and aid in verifying correct orientation within the cell.

2.2.2.1.6 Spacers

Seven spacers are provided, at approximately 18 inch intervals, to give a positive contact support for the fuel bundle rods and optimum rod spacing. The spacers provide the lateral support needed to suppress rod vibration by using inconel-x springs which hold the rods in place within the spacer matrix.

2.2.2.2 Control Rods (Figure 2.2-3)

Bottom entry control rods used in BWR's perform the dual function of power shaping and reactivity control. The main structural members of a control rod are fabricated from stainless steel and consist of a top handle casting, a bottom casting, a vertical cruciform post, and four U-shaped sheaths.

The end castings and center post are welded into a single skeletal structure. The U-shaped sheaths are spot welded to the center post and end castings to form a structurally rigid housing that contains the absorber rods.

Movement of the control rods is accomplished via the Control Rod Drive System, (Section 2.3).

2.2.2.2.1 Control Rod Blade

The control rod blade (Figure 2.2-3) consists of a sheathed, cruciform array of vertical neutron absorber rods with a blade span of 9.75 inches. The sheaths extend the full length of the control rod element and provide a continuous blade surface. A series of holes in the blade sheath permit reactor coolant to circulate freely around the absorber rods to provide cooling.

2.2.2.2.2 Absorber Rods

The absorber rods are small stainless steel tubes filled with boron carbide (B_4C) powder, vibratory compacted to 70% theoretical density. A total of 84 absorber rods, 21 in each U-shaped sheath, are contained in each control rod blade. To prevent excessive void regions, caused by settling of the boron carbide powder, stainless steel balls are spaced at 16 inch intervals within the absorber tubes. The balls are restricted in movement by spherical crimps in the tube walls. The steel balls allow gases to migrate within the tube but restrict B_4C movement to 16 inch sections.

2.2.2.2.3 Control Rod Rollers

Control rod alignment is accomplished with the use of eight spherical rollers, four of which are contained in each end casting. The four spherical rollers affixed to the top end casting guide the control rod blade between the four fuel assemblies by rolling on the fuel channels. The lower end of the control rod is laterally guided by four spherical rollers that bear on the inner surface of the control rod guide tube. The rollers are integral parts of the end castings and provide reduced resistance to movement of the control rod.

2.2.2.2.4 Control Rod Velocity Limiter (Figure 2.2-4)

The control rod velocity limiter is an integral part of the bottom end casting and is one of the engineered safeguards. The velocity limiter is designed to limit the control rod velocity in the event of a rod drop accident in order to limit the rate at which reactivity addition to the core can occur. The velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston inside the control rod guide tube.

The velocity limiter is provided with a streamlined profile in the scram or upward direction thus limiting control rod velocity only during a rod drop. When the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. However, in the dropout direction, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because the water is jetted into the annulus, a severe turbulence is created, slowing the descent of the control rod.

2.2.2.2.5 Control Rod Coupling

The control rod coupling (Figure 2.2-4) consists of a female socket, which is an integral part of the control rod bottom casting, and a spring loaded male lock plug. The upper end of the control rod drive mechanism contains a multifingered male coupling spud that fits into the female socket and is held in place by the spring loaded plug.

To couple the control rod to its drive mechanism it is only necessary to raise the drive slightly. The male coupling spud enters the socket and pushes the spring loaded male plug up. Once the coupling spud is in the correct vertical position, the fingers expand, allowing the male lock plug to drop back down into place.

2.2.3 System Features

A short discussion of system features is given in the paragraphs which follow.

2.2.3.1 Fuel Design Considerations

The fuel assembly is designed to ensure, in conjunction with core nuclear characteristics, core thermal and hydraulic characteristics, and the instrumentation and protection system, that fuel failure will not result in the release of radioactive materials in excess of guideline values in 10 CFR 20, 50, and 100.

2.2.3.2 Control Rod Design Considerations

The control rods are designed to prevent deformation that could inhibit their motion and have sufficient mechanical strength to prevent displacement of their reactivity control material.

2.2.3.3 Fuel Assembly Orientation

The fuel assembly must be oriented properly (Figure 2.2-5) in each fuel cell to ensure proper fuel bundle power distribution is achieved due to the enrichment loadings. There are five ways in which fuel assembly orientation can be verified: the channel fasteners, located at one corner of each fuel assembly, should be adjacent to the center of the fuel cell; the identification lug on the fuel assembly handle should point toward the center of the fuel cell and control rod; the channel spacer buttons should be adjacent to the control rod passage area and face one another; the assembly identification numbers located on the fuel assembly handles, should be all readable from the direction of the center of the fuel cell; and cell-to-cell duplication formed by bail handle geometric (square) pattern should occur throughout the core.

2.2.3.4 Bottom Entry Control Rods

Bottom entry control rods are used for several reasons. In a BWR, there is a large percentage of voids in the upper part of the core which significantly reduces the power in this area. If control rods entered from the top of the core, those rods, which are partially inserted, would severely depress the flux in the upper part of the core. Less time is required during refueling outages to remove and reinstall the reactor vessel head since control rod drives are not a factor as they are in PWR refueling operations. Internal moisture removal and steam separation can be more easily accomplished without interference from any top mounted control rods.

2.2.4 System Interfaces

Operation of the reactor fuel is related either directly or indirectly with almost all of the reactor plant systems. Some of the systems which relate the most directly with the Fuel and Control Rods System are discussed in the paragraphs which follow.

2.2.4.1 Reactor Vessel System (Section 2.1)

The Reactor Vessel System houses the reactor core, supports and aligns the fuel, and provides the water circulation flow paths to distribute coolant to the fuel.

2.2.4.2 Control Rod Drive System (Section 2.3)

The Control Rod Drive System provides the means by which the control rods are positioned within the reactor core.

2.2.5 Summary

Classification: Safety related system

Purpose:

The fuel: Generates energy from the nuclear fission reaction to provide heat for steam generation.

The control rods:
1. Control reactor power level
2. Control reactor power distribution
3. Provide emergency shutdown capability.

Components:

The components of the fuel are: lower tie plate, upper tie plate, fueled tie rods, standard fuel rods, water rods, fuel spacers, fuel channel, and fuel pellets.

The components of the control rod are: upper end casting, center post, lower end casting, absorber rods, blade sheath, velocity limiter, coupling device, and alignment rollers

System Interfaces:

Reactor Vessel System
Control Rod Drive System

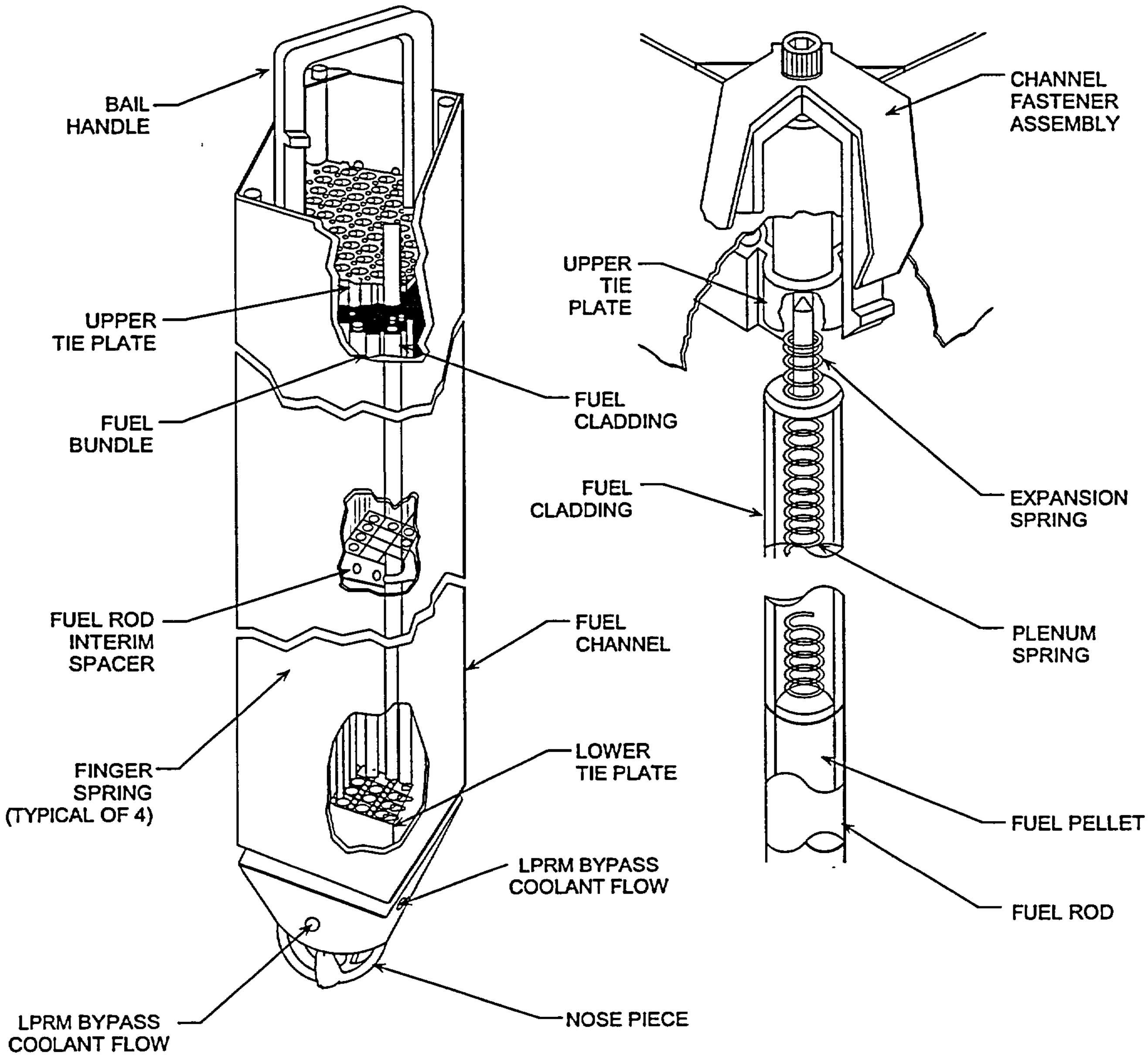

FIGURE 2.2-1 FUEL ASSEMBLY

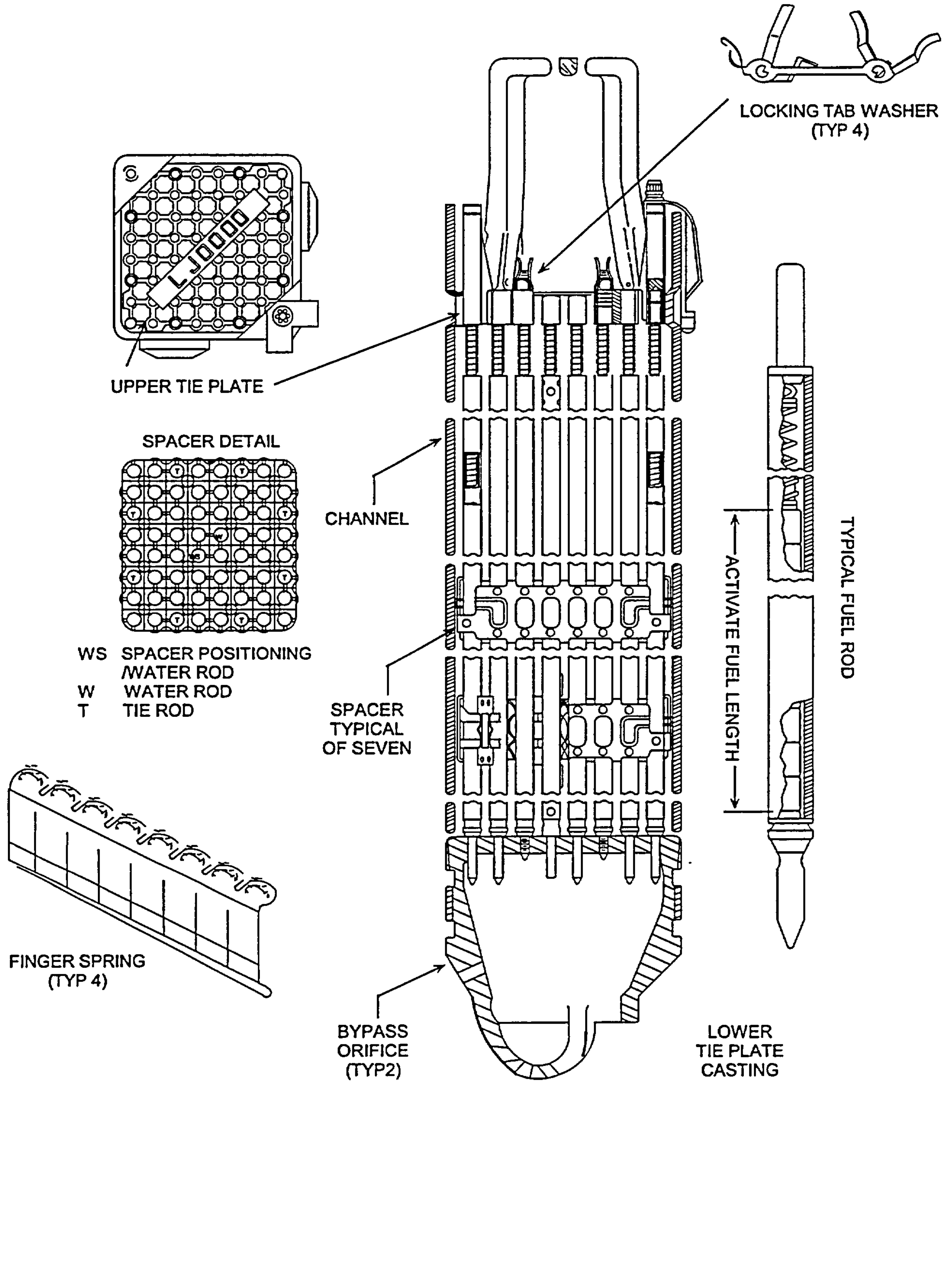

FIGURE 2.2-2 FUEL ASSEMBLY DETAILS

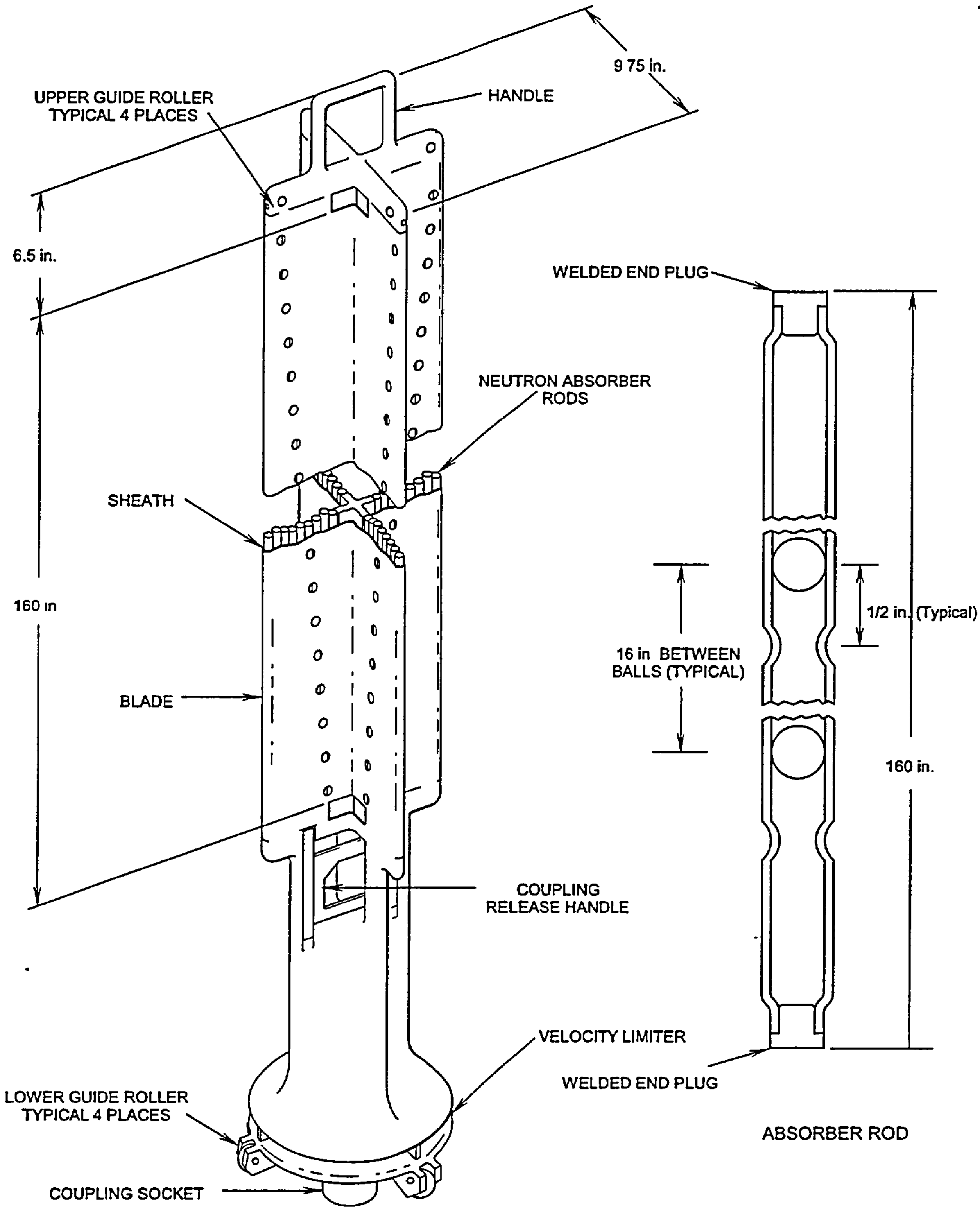

Figure 2.2-3 Control Rod

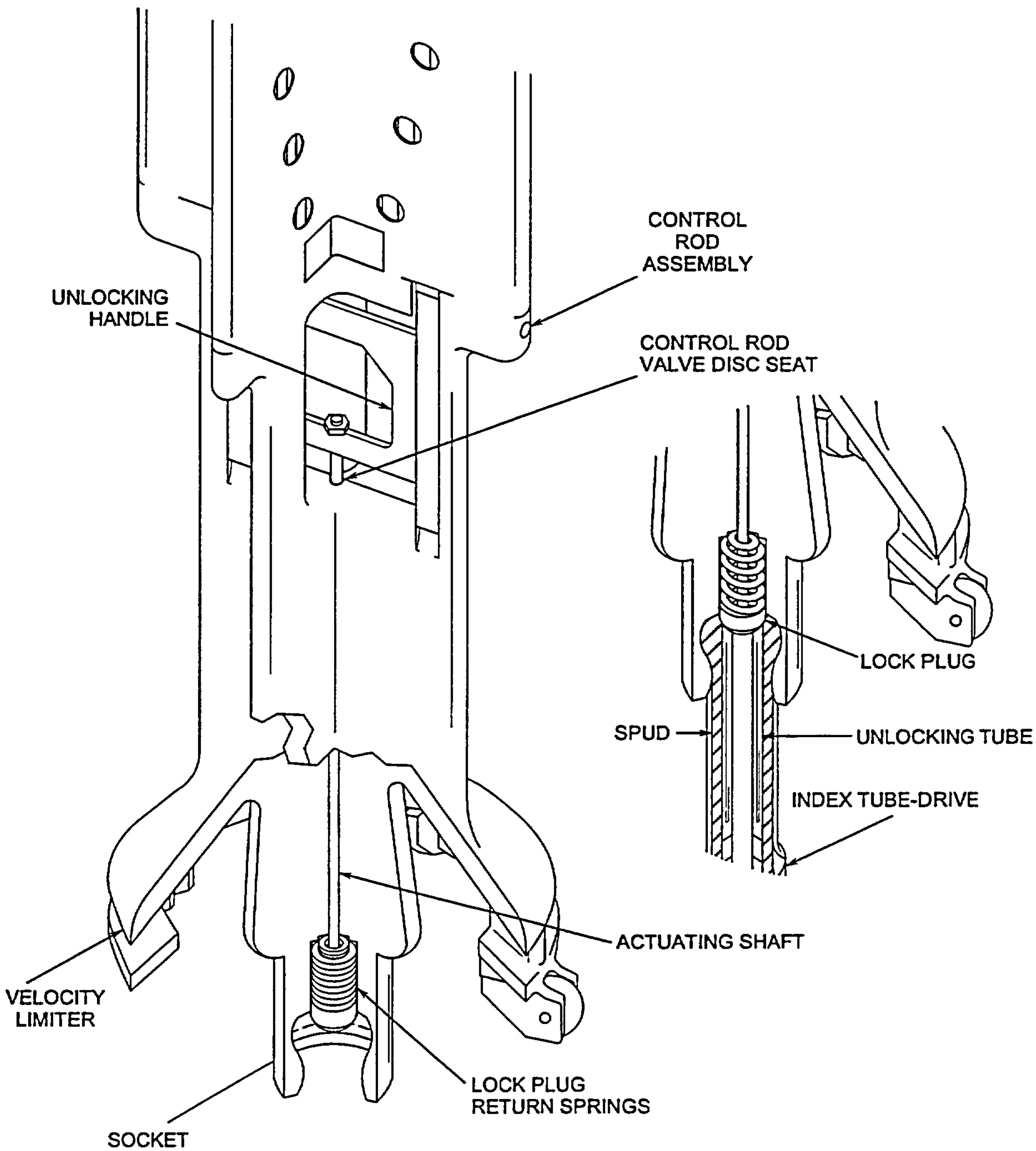

Figure 2.2-4 Control Rod Velocity Limiter and Coupling

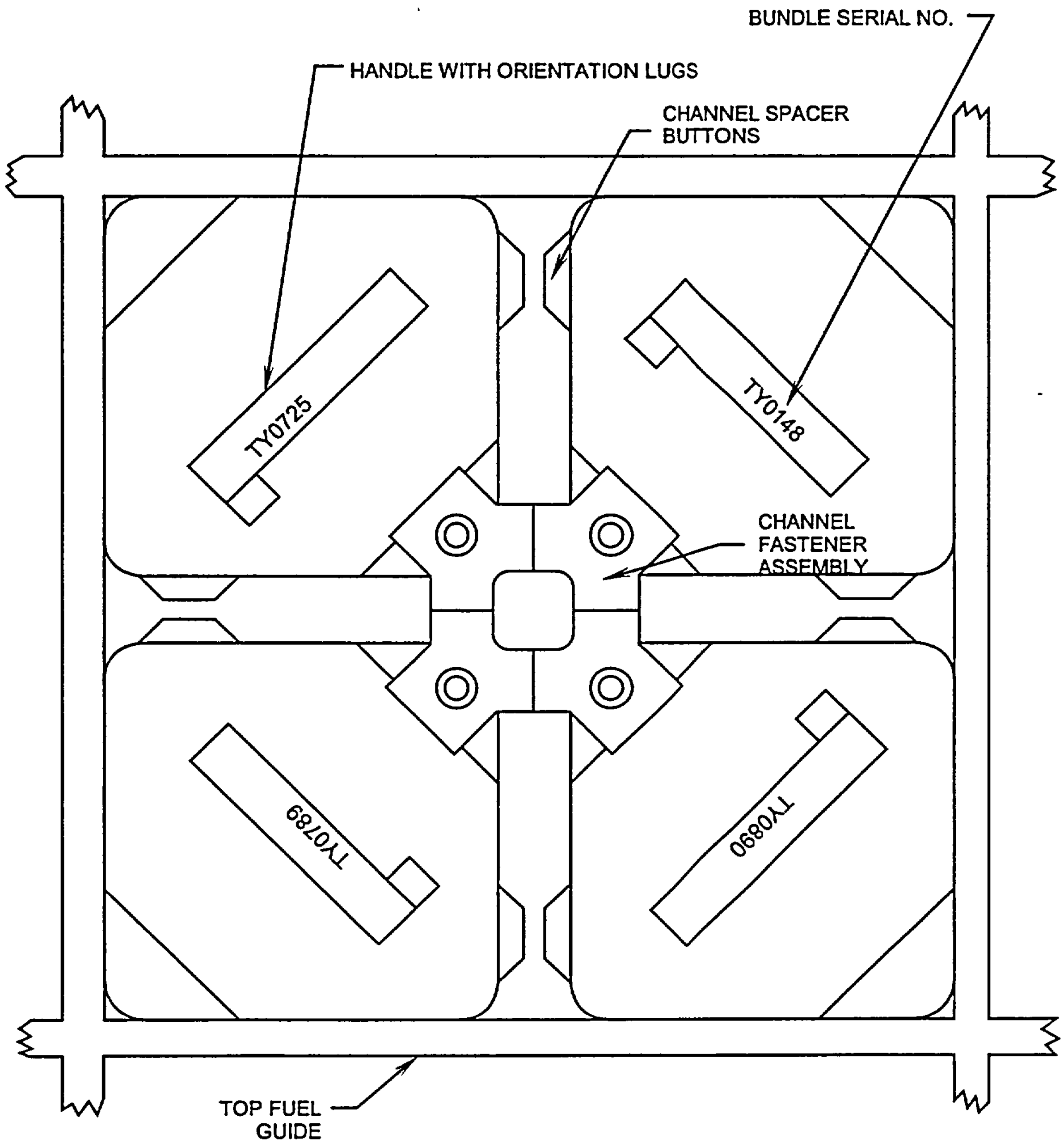

Figure 2.2-5 Fuel Assembly Orientation

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2.3 CONTROL ROD DRIVE SYSTEM

The purposes of the Control Rod Drive (CRD) System are to position control rods to:

1. Change reactor power
2. Rapidly shut down the reactor.

The Control Rod Drive System is functionally classified as a safety related system because of the Reactor Protection System scram function. Portions of the Control Rod Drive System not related to the scram function are functionally classified as power generation equipment.

2.3.1 System Description

The control rod drive mechanism, Figure 2.3-1, is a double acting, mechanically latched hydraulic cylinder using reactor quality water as operating fluid.

Control rod movement is accomplished by admitting water under pressure into the appropriate part of the control rod drive mechanism. The control rod drive mechanism movement is transmitted to the control rod blade which is coupled to the drive mechanism. The drive mechanism is capable of inserting or withdrawing a control rod at a slow controlled rate as well as providing scram insertion for rapid shutdown of the reactor.

The Control Rod Drive Hydraulic System provides the hydraulic fluid (water) for normal insertion and withdrawal of control rods. Additionally, the Control Rod Drive Hydraulic System provides cooling water for the control rod drive mechanisms and recirculation pump seals and maintains a source of stored energy for the scram function. Figure 2.3-1 shows the basic hydraulic water flow path which consists of centrifugal pumps, filters, control valves and headers which supply hydraulic fluid to the 137 control rod drive mechanisms.

2.3.2 Component Description

The major components of the CRD system are discussed in the paragraphs which follow.

2.3.2.1 CRD Pumps

The Control Rod Drive System has two fully redundant pumps, each with a rated capacity of 200 gpm at 1600 psig. The two 100% capacity pumps are motor driven, multistage centrifugal pumps. One pump is normally in service, taking suction from the condensate storage tank and providing approximately 50 to 60 gpm flow rate. The other pump is in standby.

2.3.2.2 Pump Discharge Filters

The control rod drive pump discharge filters prevent foreign material from entering the hydraulic control units. The two 100% capacity filters are redundant, with one in operation and the other manually valved out of service.

2.3.2.3 Recirculation Pump Seal Purge Supply

The control rod drive system supplies a regulated 3 to 5 gpm flow to each of the recirculation pumps (Section 2.4). The cool, clean, high quality water minimizes the crud buildup in the recirculation pump seals thus increasing seal life.

2.3.2.4 Charging Water Header

The charging water header supplies the high pressure water required for charging the water side of the scram accumulators on the hydraulic control units.

2.3.2.5 Flow Control Station

The flow control station consists of a flow element, a transmitter, a flow controller, and two 100% capacity air operated flow control valves.

The flow controller establishes a flow setpoint, set by the operator to maintain the proper cooling water header flow. The flow control valve will automatically react to deviations in the flow, sensed by the flow element, and make necessary adjustments in flow from the drive water pump. Only one flow control valve is in service with the other manually valved out.

2.3.2.6 Drive Water Pressure Control Station

The drive water pressure control station consists of a motor operated valve and two sets of stabilizing valves.

The motor operated pressure control valve is throttled to maintain approximately 260 psid above reactor pressure in the drive water header. Drive water, as its name implies, is the operating hydraulic fluid used for normal movement of the control rod drive mechanism and its associated control rod into or out of the core. For this reason, there is flow in the drive water header only during drive movement.

Two duplicate sets of stabilizing valves are installed in parallel with one set in operation and one in standby. Each set consists of two solenoid operated valves. The stabilizing valves bypass a regulated flow around the motor operated pressure control valve equivalent to the flow needed for both insertion and withdrawal of a control rod.

When a control rod is moved, the Reactor Manual Control System (Section 7.1) signals one of the stabilizing valves to close and its flow is diverted to the drive water header. One of the solenoid operated valves is used for drive insertion and the other for drive withdrawal. Thus flow remains constant in the system.

2.3.2.7 Cooling Water Header

The Control Rod Drive System cooling water header supplies cooling water, .25 gpm/drive, to each control rod drive mechanism. The cooling water being supplied to the drive mechanisms is used to cool the drives when they are stationary, prolonging the life of the seals on the drive piston.

The cooling water pressure is maintained slightly above reactor pressure by the action of the automatic flow control valve and the pressure drop across the drive water pressure control station.

2.3.2.8 Directional Control Valves

The purpose of the directional control valves is to direct drive and exhaust water for control rod movement. The Reactor Manual Control System (Section 7.1), upon command, provides the proper sequencing and duration of signals used to operate the directional control valves.

2.3.2.9 Scram Inlet and Outlet Valves

The scram inlet and outlet valves control the flow of water necessary for rapid rod insertion. The scram valves are held closed during reactor operation by instrument air pressure being applied to the tops of their actuators. The scram valves open by removing the air pressure and allowing spring force to push the valve open. Control of the air supplied to the scram valves is accomplished with the use of scram solenoid valves. The scram solenoid valves are normally energized and controlled by the Reactor Protection System (Section 7.3).

2.3.2.10 Scram Accumulators

The scram accumulator is a piston type water accumulator pressurized by a volume of N_2 gas in a N_2 cylinder. The accumulators and their instrumentation occupy the lower part of the hydraulic control units. The piston in the scram accumulator forms a barrier between the high pressure N_2 gas used as the source of stored energy and the water used to initiate the scram. Under normal plant operating conditions the piston is in the full down position. The control rod drive pump continuously pressurizes the scram accumulators through the charging water header.

2.3.2.11 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each Hydraulic Control Unit and drains into two independent, redundant instrument volumes. The scram discharge volume is sized to receive and contain approximately twice the volume of water discharged by all 137 control rod drive mechanisms during a scram, independent of the instrument volumes. During normal operation the scram discharge volume is empty, and vented to atmosphere by air operated globe valves that automatically close on a scram signal from the Reactor Protection System.

2.3.2.12 Hydraulic Control Units (HCU)

The HCU, shown in Figure 2.3-2, includes all the hydraulic, electrical and pneumatic equipment necessary to move one control rod drive mechanism during normal or scram operation. There are 137 HCU's, divided into approximately two equal sets, which are located in the reactor building on each side of the drywell.

Each HCU performs three specific functions:

1. Stores energy (accumulators) and contains the valving necessary (scram inlet and outlet valves) to permit the control rod to scram.

2. Contains valving necessary for normal movement (solenoid operated directional control valves).

3. Provides a cooling water flow path to the control rod drive mechanism.

2.3.2.13 Control Rod Drive Mechanism

The control rod drive mechanism (CRDM) is shown functionally in Figure 2.3-1. Figure 2.3-3, illustrates the normal latch condition of the mechanism and the cooling water flow path.

The CRDM is a double acting, mechanically latched hydraulic cylinder assembly using reactor quality water as its operating fluid for normal and scram control rod operation. The CRDM is capable of inserting or withdrawing a control rod at a slow, controlled rate, as well as providing a rapid insertion in the event of an off normal condition. A locking mechanism in the CRDM permits the control rod to be positioned at 6 inch increments of stroke and to be held in a set position for indefinite periods of time.

The basic parts of a CRDM consists of a control rod drive housing, an outer tube, an inner cylinder, an index tube, a piston tube, a drive piston, a collet locking mechanism, and a position indicating tube.

Control rod position is obtained from reed switches mounted inside of the position indication probe, that open or close during rod movement. The reed switches are provided at three inch increments of drive piston travel. Since a notch is six inches, indication is available for each half notch of rod travel. Even numbered readouts, 00 to 48, are provided at each latched drive position.

Odd numbered readouts, 01 to 47, indicate the midpoints between the latched positions.

The outer tube, inner cylinder and flange is a single piece unit. The flange provides the means of mounting the CRDM to the control rod drive housing flange. The outer tube and inner cylinder form an annulus through which water is applied to the collet locking mechanism to unlock the index tube.

The drive piston and index tube provide the driving link with the control rod as well as notches for the collet locking mechanism fingers. The notches are machined to engage the collet fingers, providing 25 increments (at 6 inch intervals) at which a control rod may be positioned.

The collet locking mechanism is contained in the upper portion of the outer tube and ensures the index tube is locked to hold the control rod at a selected position in the reactor core.

2.3.2.14 Control Rod Drive Housing Support

The control rod drive housing support network is an engineered safety feature designed to limit the ejection of a control rod to less than 3 inches in the unlikely event of a control rod drive housing failure with the reactor pressurized. This network consists of support beams on the inside of the reactor pedestal, with hanger rods and spring washers suspended from beams.

2.3.3 System Features

A short discussion of system features is given in the paragraphs which follow:

2.3.3.1 Normal Operation (Figure 2.3-3)

During normal power operation, the control rod drive system is providing cooling water flow to all of the control rod drive mechanisms and both recirculation pumps. Cooling water enters through the insert port of each CRDM and is allowed to flow between the outer tube and thermal sleeve via a cooling water orifice. The cooling water flow rate is regulated by the restricting orifice and allows approximately .25 gpm to pass. Cooling water is provided to ensure longer seal and bushing life.

Cooling water flow is normally maintained to each CRDM at all times except when a control rod is moved.

2.3.3.2 Control Rod Insert Operation

Control rod insert operation is accomplished by opening both insert directional control valves 121 and 123. Opening of the directional control valves applies drive water pressure to the under side of the drive piston and allows water from the area above the drive piston to exhaust to the exhaust header. The water displaced into the exhaust header is then returned to the reactor vessel.

The differential pressure that is created across the drive piston forces the drive mechanism into the core, inserting the control rod. With the drive moving upward, the collet fingers move outward and provide no restriction to the drive.

2.3.3.3 Control Rod Withdraw Operation

Control rod withdraw requires a brief insert signal (Section 2.3.3.2) to relieve the axial load on the collet locking mechanism. The area of the collet piston is designed to ensure unlocking is not possible when opposed by the forces of its collet spring, the weight of the drive components, and the driving pressure applied to the area above the drive piston.

Immediately following the brief insert signal a withdraw signal is applied to the drive mechanism by opening the 120 and 122 directional control solenoid valves. Drive water pressure is simultaneously applied to the collet piston and the area above the drive piston while the area below the drive piston is aligned to the exhaust header.

With the area below the drive piston aligned to the exhaust header and the area above the piston pressurized with drive water pressure, a differentional pressure is created to move the drive out of the core.

2.3.3.4 Settle Operation

At the termination of any control rod insert or withdraw signal, the Reactor Manual Control System automatically energizes and opens the withdraw directional control solenoid valve 120 for several seconds. This opens the area under the drive piston to the exhaust header, permitting the drive to settle downward to a new latched position.

2.3.3.5 Control Rod Scram Operation

When a reactor scram is initiated by the Reactor Protection System, the scram inlet valves open to admit pressurized water from the scram accumulator to the area below the drive piston, and the scram outlet valves open to vent the area above the drive piston to the scram discharge volume. The large differential pressure applied to the drive piston produces a large upward force on the control rod drive, giving it a high initial acceleration and provides an excess amount of force to overcome friction forces within the control rod drive mechanism.

The drive internals provide an additional pressure source to back up or aid accumulator pressure during a scram. The ball check valve, located in the insert port, will unseat and maintain the pressure under the drive piston at reactor pressure via internal ports in the drive. At a reactor pressure greater than 800 psig, reactor pressure alone is capable of scramming the drive. At reactor pressures less than 800 psig, the accumulator is necessary to scram the drive.

During and following a scram condition, a large flow condition exists to the charging water header. The large flow signal closes the flow control valve thus directing all water to the charging water header for accumulator charging.

2.3.4 System Interfaces

In addition to the Reactor Manual Control System and Reactor Protection System, the following systems interface with the CRD System.

2.3.4.1 Recirculation System (Section 2.4)

The CRD System supplies seal water to each recirculation pump seal cavity.

2.3.4.2 Condensate and Feedwater System (Section 2.6)

The condensate and feedwater system is the source of water used by the CRD system.

2.3.4.3 Radwaste System

The scram discharge instrument volumes drain to the Clean Radwaste System.

2.3.4.4 Service and Instrument Air System

Instrument air from the Service and Instrument Air System supplies high quality air to the CRD system air operated components.

2.3.4.5 Fuel and Control Rods

The control rods are positioned in the core by the CRD system.

2.3.4.6 Reactor Water Cleanup System (Section 2.8)

The control rod drive system has the capability to supply water directly to the reactor vessel via the reactor water cleanup return line.

2.3.5 Summary

Classification:

All components of the CRD System required for a scram constitute a safety system.

Purpose:

To position control rods to:.

1. Change reactor power
2. Rapidly shutdown the reactor

Components:
The major components of the Control Rod Drive System are: hydraulic supply water, flow control station, scram discharge volume,CRDM,CRD support structure,HCU's.

System Interfaces:
Condensate and Feedwater System, Recirculation System, Radwaste System, Service and Instrument Air System, Fuel and Control Rods System, Reactor Protection System, Reactor Manual Control System.

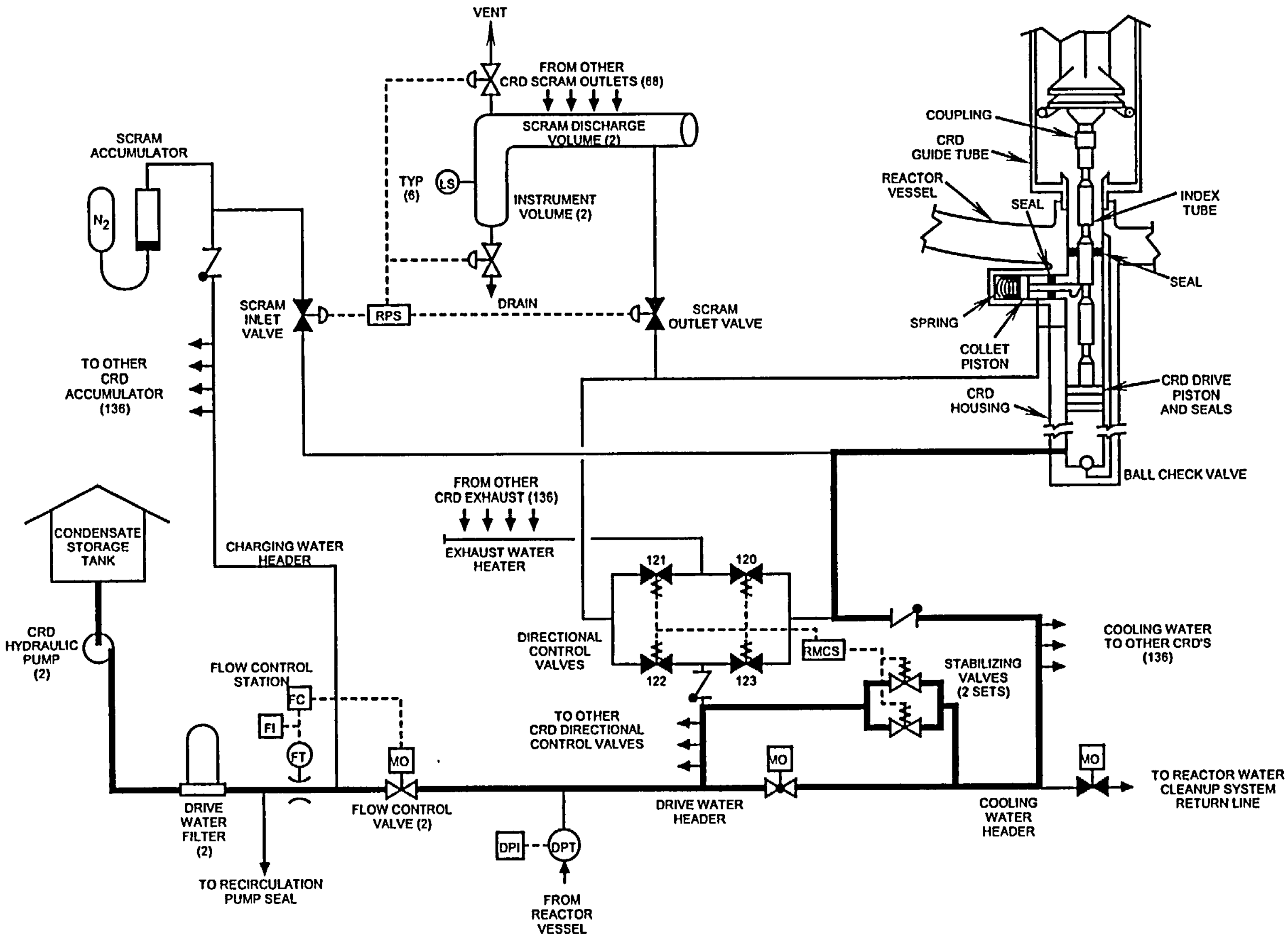

Figure 2.3-1 Control Rod Drive Hydraulic System

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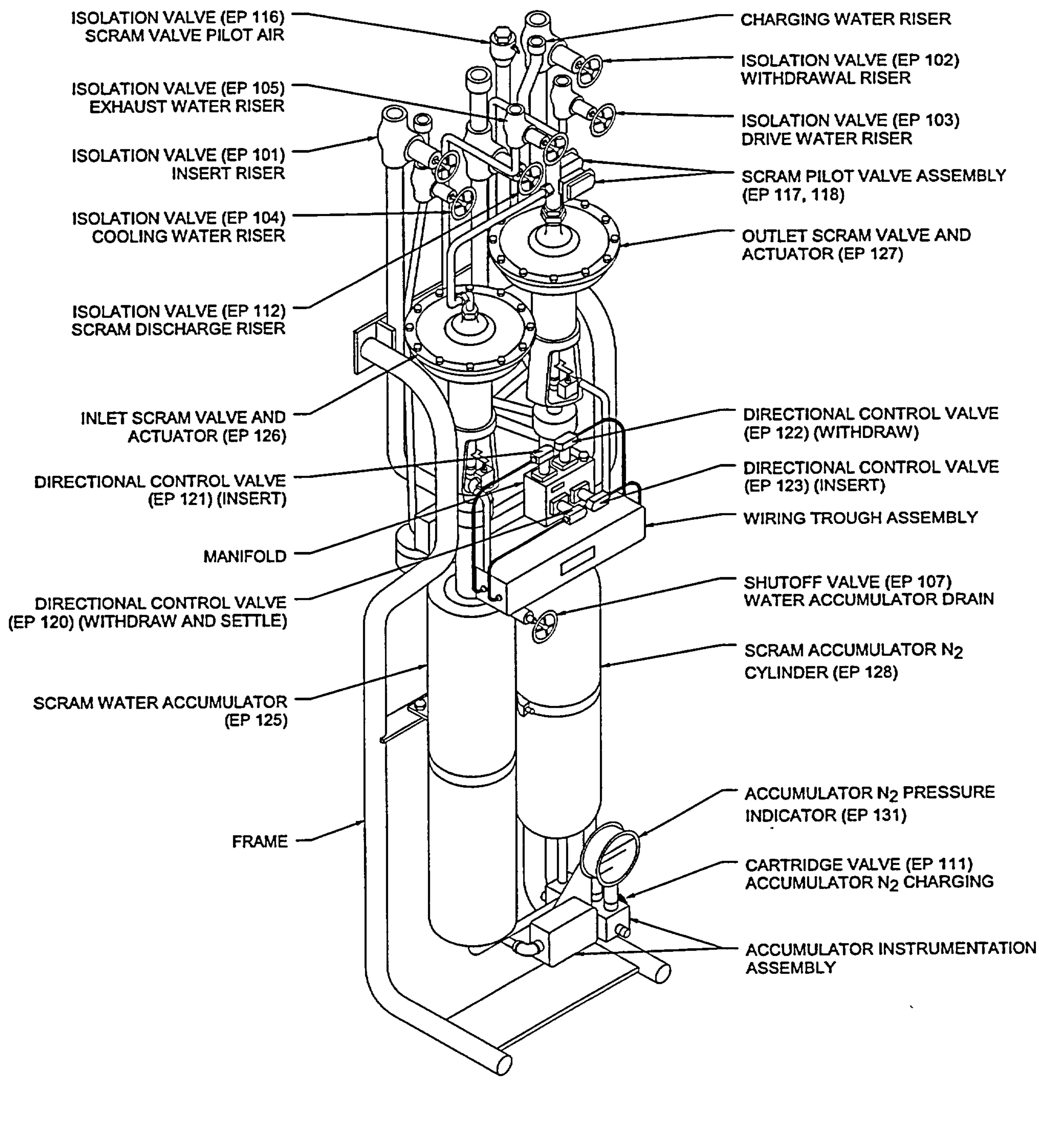

Figure 2.3-2 Hydraulic Control Unit

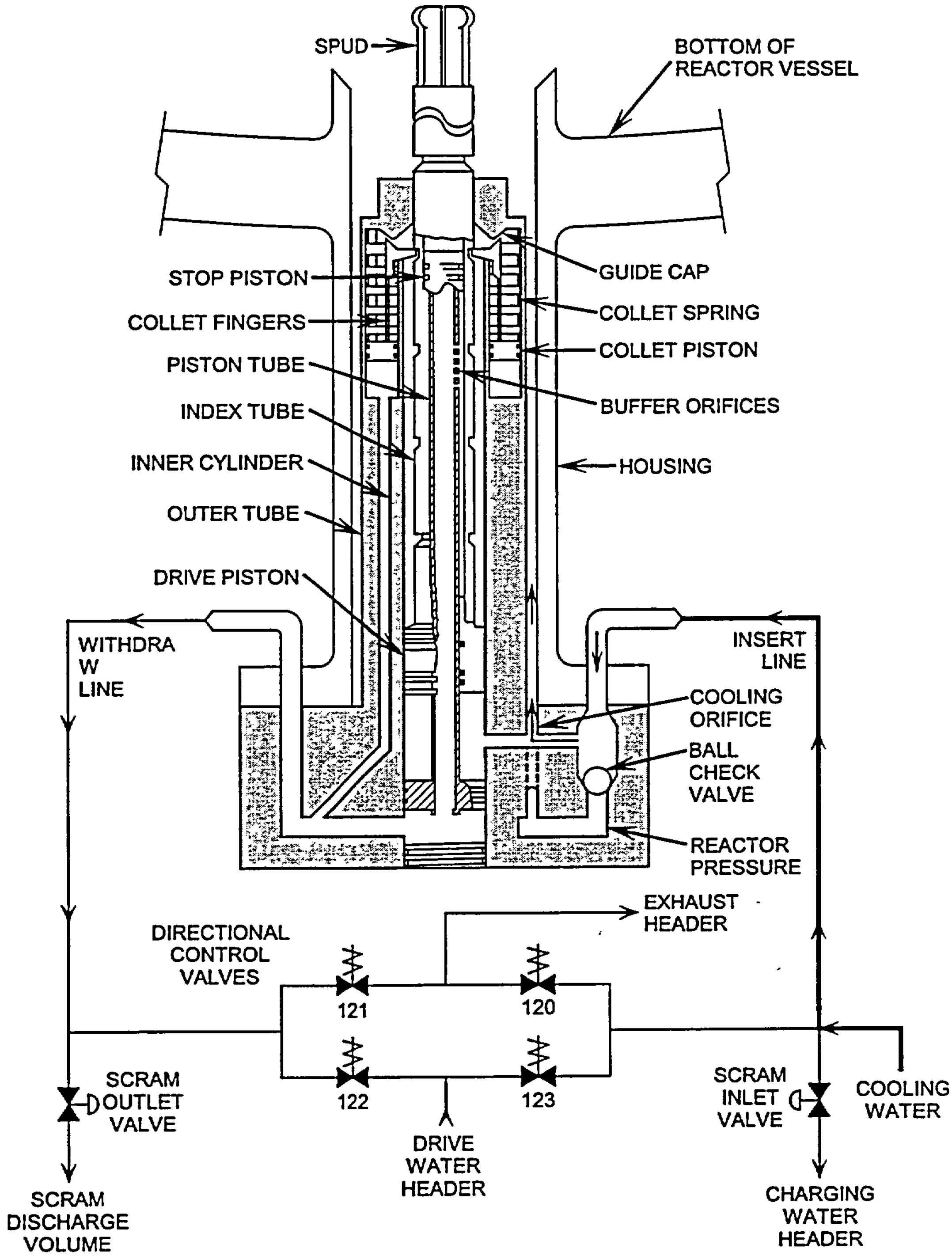

Figure 2.3-3 CRD

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2.4 RECIRCULATION SYSTEM

The purpose of the Recirculation System is to provide forced circulation of water through the reactor core, permitting higher reactor power than with natural circulation.

The functional classification of the Recirculation System is that of a power generation system.

2.4.1 System Description

The Recirculation System provides the driving force to circulate coolant through the reactor core. The system consists of two piping loops external to the reactor vessel and 20 jet pumps which are internal to the reactor vessel. Each loop has a suction isolation valve, a recirculation pump, a discharge isolation valve, instrumentation, and piping connecting to the reactor vessel. The Recirculation System is shown in Figure 2.4-1.

The variable speed recirculation pumps take suction from the reactor vessel annulus region and provide flow to the jet pump riser pipes through the reactor vessel shell. Internal piping directs the driving flow to the jet pumps. The jet pumps induce additional water from the reactor vessel annulus region into the flowpath and force it to flow through the core.

2.4.2 Component Description

The major components of the Recirculation System are discussed in the paragraphs which follow.

2.4.2.1 Suction Valves

There is a suction valve in each recirculation loop between the reactor vessel penetration and the recirculation pump. These suction valves are motor operated isolation valves which are normally open but which can be closed to isolate the recirculation pump for maintenance. There are no automatic closure conditions associated with these valves.

2.4.2.2 Recirculation Pumps

There is a recirculation pump in each recirculation loop between the suction and discharge valves. These pumps are vertical, single stage, centrifugal pumps driven by variable speed electric motors. The pumps provide the driving flow to the jet pumps and have a rated flow of 45,200 gallons per minute each. The speed of the recirculation pumps, and hence the system flowrate, is controlled by the Recirculation Flow Control System (Section 7.2).

Each recirculation pump is equipped with a shaft seal assembly, shown in Figure 2.4-2, to prevent leakage of recirculation water along the shaft which connects the recirculation pump motor to the recirculation pump. The shaft seal assembly consists of two individual, 100% capacity, mechanical seals built into a single cartridge. Although each seal can individually seal against full primary system pressure, the normal pressure drop across each seal is about half of that value, or 500 psid. This pressure differential is provided by the control pressure breakdown orifices. In the event that either seal should fail, the remaining seal can withstand the full pressure drop from rated system pressure to atmospheric and still maintain seal integrity. In the event that both seals should fail, there is a breakdown bushing in the recirculation pump casing to limit water leakage along the shaft to a nominal value of 60 gallons per minute.

Provisions are made for monitoring the pressure drop across each individual seal as well as the controlled seal leakage flow. Various control room alarms indicate improper seal performance.

Seal purge water, supplied from the Control Rod Drive System (Section 2.3), keeps the recirculation pump seals clean and cool. Approximately 3 gpm of cool, filtered water is injected between the number one seal and the breakdown bushing. The seal purge substantially reduces the possibility of the seals being damaged by dirt from water supplied by the recirculation pump water.

2.4.2.3 Discharge Valves

Each recirculation loop contains a motor operated discharge valve located between the recirculation pump and the loop flow measurement device. The valve is remotely operated from the control room using an open/close control switch. The discharge valve is sealed-in to close and throttle to open. The discharge valves are automatically jogged open on a pump startup by the Recirculation Flow Control System (Section 7.2). Additionally, discharge valves close as part of the automatic initiation sequence for the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System (Section 10.4) to provide an emergency core cooling water flowpath to the reactor vessel.

2.4.2.4 Jet Pumps

There is a bank of 10 jet pumps associated with each of the external recirculation loops. All jet pumps are located in the reactor vessel annulus region between the inner vessel wall and the core shroud. The jet pumps are provided to increase the total core flow while minimizing the flow external to the reactor vessel (recirculation loop flows).

Each jet pump has a converging nozzle through which the driving flow passes. This creates a high velocity and relatively low pressure condition at the jet pump suction. This low pressure condition creates additional flow from the vessel annulus, called induced flow, through the jet pump. The combined driving flow and induced flow mix in the mixer section of the jet pump and then pass through the diffuser section. The diffuser section increases the flow area which decreases fluid velocity and increases pressure. During full power operation approximately one-third of the total core flow comes from the discharge of the recirculation pumps while the remaining two-thirds is induced by the jet pumps.

2.4.3 System Features

A short discussion of system features is given in the paragraphs which follow.

2.4.3.1 System Operation

The recirculation pumps are started prior to control rod withdrawal for a plant startup and are both run at the same speed during power operation of the reactor. Should one recirculation loop become inoperable, plant operation may continue for a specified time as limited by the plant Technical Specifications. This is known as single recirculation loop operation.

Should both recirculation loops become inoperable, plant operation may continue for another specified time period if permitted by the plant Technical Specifications. This is a condition known as natural circulation. Even though there is not external recirculation loop drive flow, there still is natural circulation flow through the jet pump suctions. The natural circulation phenomenon occurs within the reactor vessel because of coolant density differences between the reactor core region and the vessel annulus.

2.4.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

2.4.4.1 Reactor Vessel System (Section 2.1)

The jet pumps are mounted in the reactor vessel annulus area; the jet pump riser pipes represent several vessel penetrations; the recirculation suction and discharge lines represent the largest vessel penetrations.

2.4.4.2 Recirculation Flow Control System (Section 7.2)

The Recirculation Flow Control System controls the speed of the recirculation pumps by adjusting the electrical frequency and voltage supplied to the pump motors.

2.4.4.3 Residual Heat Removal System (Section 10.4)

Reactor water is supplied to the Residual Heat Removal System from one of the recirculation suction lines for the shutdown cooling/head spray mode of RHR System operation. After the reactor water has been cooled by the heat exchangers in the RHR System, it is returned to the reactor vessel via RHR System penetrations in the discharge piping of both recirculation loops. Additionally the low pressure coolant injection (LPCI) mode of the RHR System supplies emergency core cooling water, under accident conditions, to the reactor via the same recirculation loop discharge penetrations.

2.4.4.4 Reactor Water Cleanup System (Section 2.8)

Reactor water is supplied to the Reactor Water Cleanup System for purification from one of the recirculation loop suction lines.

2.4.4.5 Neutron Monitoring System (Chapter 5.0)

Recirculation loop flow signals are sent to the Neutron Monitoring System for use in protective trip functions when the reactor mode switch is in the RUN position.

2.4.4.6 Control Rod Drive System (Section 2.3)

The recirculation pump seals are supplied with cool clean seal purge water from the Control Rod Drive System.

2.4.5 Summary

Classification:

Power generation system

Purpose:

To provide forced circulation of water through the reactor core, permitting higher reactor power than with natural circulation.

Components:

Suction valves; recirculation pumps;discharge valves; jet pumps.

System Interfaces Recirculation Flow Control System;Residual Heat Removal System;Reactor Water Cleanup System; Neutron Monitoring System; Control Rod Drive System; Reactor Vessel System.

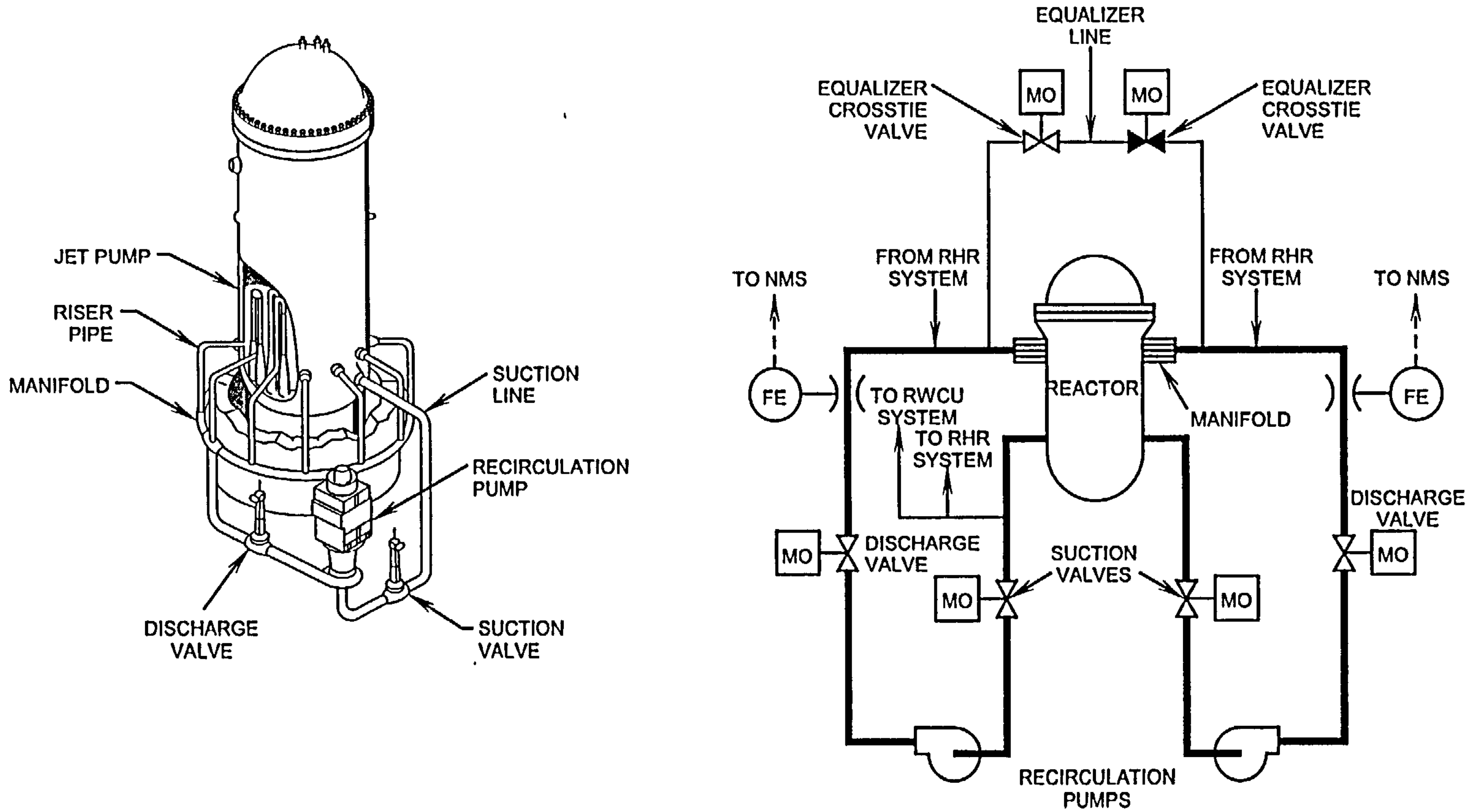

Figure 2.4-1 Recirculation System

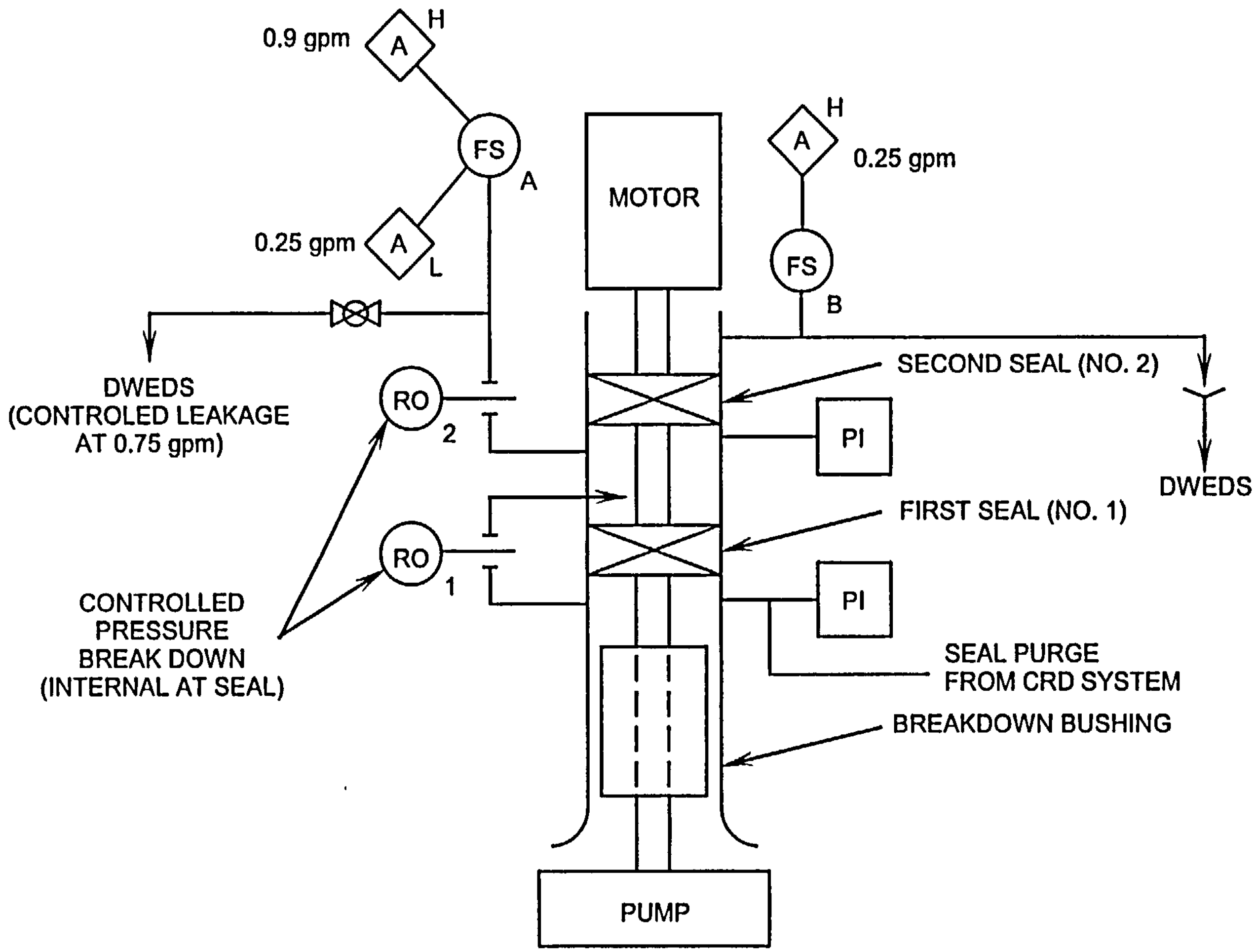

Figure 2.4-2 Shaft Seal Assembly

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2.5 MAIN STEAM SYSTEM

The purposes of the Main Steam System are:

1. To direct steam from the reactor vessel to the main turbine and other steam loads.

2. To provide overpressure protection for the Reactor Coolant System.

The functional classification of the Main Steam System is that of a power generation system. The main steam system does, however, contain three components which are engineered safety features (ESF). These ESFs are the main steam isolation valves, the main steam line flow restrictors, and the safety/relief valves.

2.5.1 System Description

The flow path for the Main Steam System is shown on Figure 2.5-1. Within the drywell, steam is transmitted from the reactor vessel to the inboard main steam isolation valves via 4 steam lines. Each of the lines contain safety/relief valves, steam line flow restrictors, drain line connection and inboard isolation valves. Also provided from the steam lines is a reliable source of steam for the High Pressure Coolant Injection System and the Reactor Core Isolation Cooling System.

Outside of the drywell, the steam progresses through outboard isolation valves to an equalizing header where all 4 of the steam lines terminate. From the equalizing header steam is routed to the main turbine and a number of auxiliary support systems.

2.5.2 Component Description

The major components of the Main Steam System are discussed in the paragraphs which follow.

2.5.2.1 Safety/Relief Valves (Figure 2.5-2)

The Safety/Relief valves (11) provide overpressure protection for the reactor vessel and associated piping systems. In addition a selected number of safety/relief valves are used by the Automatic Depressurization System (Section 10.2).

All of the safety/relief valves are mounted on a horizontal portion of the main steam lines inside the drywell. They are dual actuating type valves and discharge to the suppression pool. Actuation is accomplished by either; (1) high system pressure (safety mode); (2) remotely by operator action; (3) the ADS logic .

In the safety mode, the safety/relief valves are actuated by high pressure steam positioning a pilot valve against setpoint spring pressure to depressurize the top of the main valve piston. The removal of steam on top of the main valve piston creates a differential pressure which forces the main valve open. Steam is released through the valve to the suppression pool, lowering reactor pressure.

In the relief mode of operation, air pressure is applied to an air actuator by energizing a solenoid operated valve. The air actuator mechanically positions the pilot valve, depressurizing the top of the main valve piston to cause the main valve to open. The solenoids are energized by switches located in the control room. This type of arrangement provides the control room operator with a means to operate any of the 11 safety/relief valves. Seven of the safety/relief valve solenoids can also be energized by actuation of the Automatic Depressurization System logic .

2.5.2.2 Main Steam Line Flow Restrictors

The main steam line flow restrictor is a engineered safety feature incorporated into the plant design to limit the loss of mass inventory from the reactor during a steam line break. Each flow restrictor is designed to limit steam flow to less than 200% of rated steam flow during a steam line break. The flow restrictors are also used to develop steam flow signals for use in the Feedwater Control System (Section 3.3) and Primary Containment Isolation System (Section 4.4).

2.5.2. Main Steam Isolation Valves

The Main Steam Isolation Valves (MSIV's) are provided to limit the release of radioactive materials to the environment following a steam line break outside of containment. Redundant valves are located on each main steam line, one inside and one outside the primary containment. The MSIV's are air/spring operated valves; air to open, air and/or spring to close.

2.5.2.4 Main Steam Line Drains

The main steam line drains are provided to serve a dual function. The first function is to remove moisture from the steam lines prior to and during turbine operation. The removal of moisture helps prevent damage to the turbine blades. The steam line drains also provides the capability to equalize pressure around the MSIVs prior to opening.

2.5.2.5 Bypass Valves

Four bypass valves are used to bypass up to 30% of rated steam flow directly to the condenser when the main turbine is not capable of using the steam. The bypass valves work in conjunction with the turbine control valves to ensure a constant reactor pressure for a given reactor power level. Operation of the bypass valves is automatically controlled by the Electro Hydraulic Control System (Section 3.2).

2.5.2.6 Turbine Stop Valves

There are four turbine stop valves located in the main steam piping just upstream of turbine control valves (Figure 2.5-1). The stop valves are normally open during turbine operation with a rapid closure capability upon detection of potentially unsafe turbine conditions. The number two stop valve also contains an internal bypass which is used to equalize pressure across the stop valves prior to opening.

2.5.2.7 Turbine Control Valves

The control valves regulate the steam flow to the turbine, as controlled by the Electro Hydraulic Control (EHC) System, in order to control reactor pressure. The control valves also provide the control mechanism for rolling, synchronizing, and loading the turbine generator.

2.5.2.8 Turbine

The main turbine is an 1800 RPM, tandem compound machine, consisting of one high pressure turbine and three low pressure turbines. The steam enters the middle of the high pressure section and works its way toward each end, dissipating its energy to the turbine blades which are attached to a common shaft.

Steam exiting the high pressure turbine passes through moisture separators where entrained moisture, which could cause turbine blade damage, is removed prior to entering the low pressure turbines.

Prior to entering the low pressure turbines, the steam passes through the low pressure turbine control and stop valves which are called combined intermediate valves (CIV's) and are actually two valves in one housing. The intercept valves provide throttling to prevent turbine overspeed caused by a rapid reduction in generator load. The CIV stop valve is designed to be fully open or fully closed (open during normal operation) and provides a rapid means of isolating the low pressure turbines from their steam supply when necessary.

At different stages on the low pressure turbine are taps for extracting steam. This steam is used to preheat the feedwater going back to the reactor vessel (Section 2.6).

The exhaust of the low pressure turbine sections is directed to the condenser, where it is condensed and deaerated, then collected in a hotwell to be pumped by the Condensate and Feedwater System to the reactor vessel.

2.5.2.9 Other Steam Equipment

The Main Steam System supplies steam to a number of components within the plant. Below is a listing of some of those components and a description of those not covered elsewhere in this manual.

1. Steam Jet Air Ejector (SJAE)
2. Gland Seal System
3. Off Gas System (Section 8.1)
4. Reactor Feed Pump Turbines (Section 3.3)
5. Reactor Core Isolation Cooling System (Section 2.7)
6. High Pressure Coolant Injection System (Section 10 . 1)

2.5.2.10 Steam Jet Air Ejector (SJAE)

The Steam Jet Air Ejectors remove noncondensible gases from the condenser and provide the motive force for moving these gases through the Offgas System.

2.5.2.11 Gland Seal Steam

The steam space along the turbine shaft, between the turbine's last stage and atmosphere is sealed in order to maintain the vacuum in the main condenser and to prevent contaminated steam from leaking to atmosphere. Sealing is accomplished by means of labyrinth type shaft packing which provide a series of throttling that limit steam leakage along the rotating shaft to a minimum as it is throttled from the high pressure space to the low pressure space.

Air and steam, possibly radioactive, are removed from the seals through a steam packing exhaust system and discharged to the plant stack system via a short holdup line (primarily for N16 decay).

2.5.3 System Features

A short discussion of system features is given in the paragraphs which follow.

2.5.3.1 Main Steam Isolation Valves

To minimize the release of radioactive materials to the environment the MSIV's receive automatic closure signals from the Primary Containment Isolation System (Section 4.4). Automatic closure of all eight MSIV's is initiated under any of the following conditions:

1. Low-Low Reactor Vessel Water Level
2. High Main Steam Line Radiation
3. High Main Steam Line Flow
4. High Steam Line Area Temperature
5. Low Steam Line Pressure with the mode switch in Run.

MSIV closure, with the reactor critical, can result in a severe pressure and power increase. Because of this, MSIV closure signals the Reactor Protection System (Section 7.3) to scram the reactor. The valves are required to close in >3 but

<5 seconds to minimize the pressure/power increase and at the same time limit the release of radioactive material on a downstream line break.

2.5.3.2 Turbine Stop and Control Valve Closure

The hydraulically operated turbine stop and control valves can be tripped closed within 0.1 and 0.2 seconds respectively. Because of the large pressure and neutron flux spikes created by stop valve closure and control valve fast closure, the reactor is automatically scrammed and reactor recirculation pumps tripped if either condition occurs with power above the capacity of the bypass valves.

2.5.4 System Interfaces

The interfaces this system has with other plant systems are discussed in the paragraphs which follow.

2.5.4.1 Reactor Vessel System (Section 2.1)

The Main Steam System delivers steam from the reactor vessel to the various steam loads, vents noncondensible gases from the reactor vessel head area, and provides overpressure protection for the reactor vessel.

2.5.4.2 Condensate and Feedwater System (Section 2.6)

The Condensate and Feedwater System uses steam from the equalizing header or the moisture separators to drive the reactor feedpump turbines and extraction steam from the main turbine to heat the feedwater.

2.5.4.3 Reactor Core Isolation Cooling System (Section 2.7)

The Reactor Core Isolation Cooling System uses steam from the A steam line as the driving force for it's turbine.

2.5.4.4 High Pressure Coolant Injection System (Section10.1)

The High Pressure Coolant Injection System uses steam from the B steam line as the driving force for its turbine.

2.5.4.5 Electro Hydraulic Control System (Section 3.2)

The Electro Hydraulic Control System controls the operation of the bypass valves and turbines valves to control reactor pressure and turbine generator load.

2.5.4.6 Feedwater Control System (Section 3.3)

The Feedwater Control System uses steam flow signals from the steam line flow restrictors taps as part of the three element level control network and for indication.

2.5.4.7 Primary Containment IsolationSystem (Section 4.4)

The Primary Containment Isolation System isolates the Main Steam System when required.

2.5.4.8 Reactor Protection System (Section 7.3)

The Reactor Protection System uses MSIV and turbine stop valve position and turbine control valve fast closure signals to initiate reactor scrams.

2.5.4.9 Offgas System (Section 8.1)

The Offgas System uses main steam to drive the steam jet air ejectors and heat the recombiner preheaters.

2.5.4.10 Liquid Radwaste System (Section 8.2)

The Liquid Radwaste System uses main steam to heat the radwaste evaporator.

2.5.4.11 Automatic Depressurization System (Section 10.2)

The Automatic Depressurization System uses six of the thirteen safety/relief valves to make up one of the four emergency core cooling systems (ECCS).

2.5.5 Summary

Classification :

Power Generation System
Purpose:

To direct steam from the reactor vessel to the main turbine and other steam loads.

To direct steam, during abnormal conditions, to certain safety related systems and provide overpressure protection for the reactor coolant system boundary.

Components:

The components which makeup the main steam system are; safety/relief valves, flow restrictors, MSIV's, BPV's, turbine stop and control valves, turbine moisture separators, CIV's, steam seals, SJAE.

System Interfaces:

Reactor Vessel System; Condensate and FeedwaterSystem; Reactor Core Isolation Cooling System; High Pressure Coolant Injection System; Electro Hydraulic Control System; Feedwater Control System; Reactor Protection System; Offgas System; Liquid Radwaste System; Automatic Depressurization System,Primary Containment Isolation System.

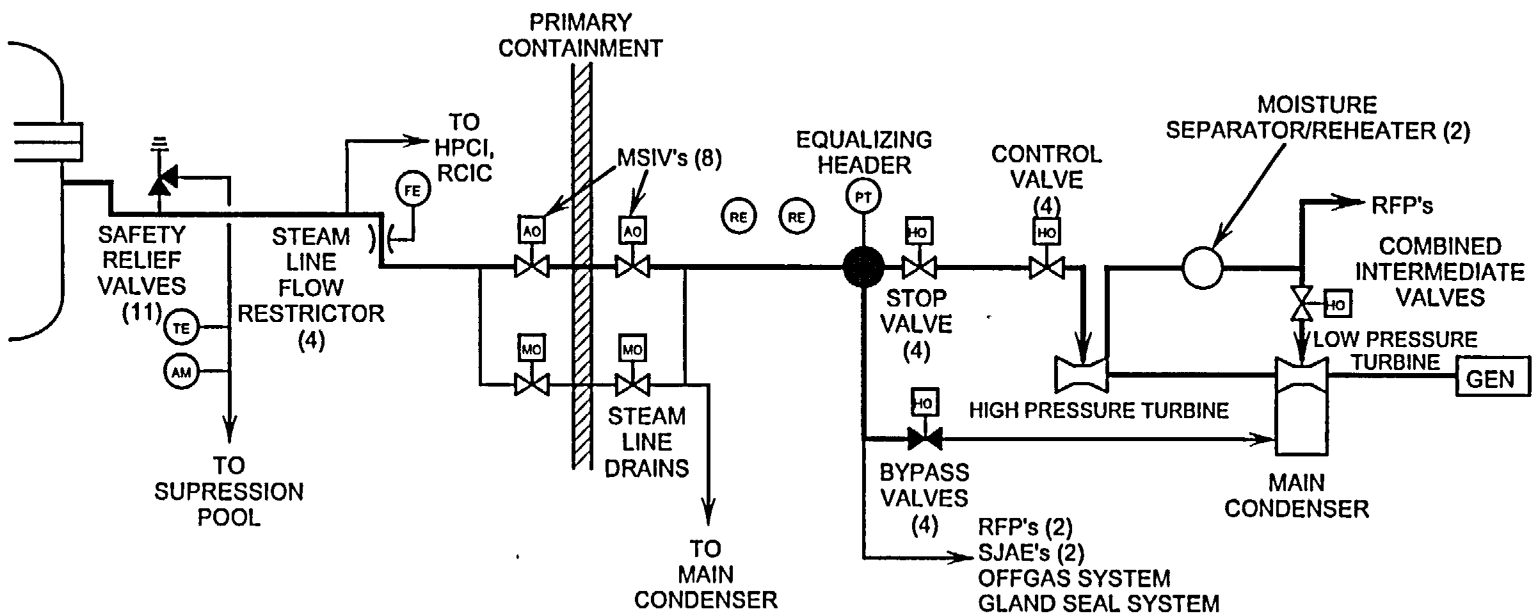

FIGURE 2.5-1 MAIN STEAM SINGLE LINE DIAGRAM

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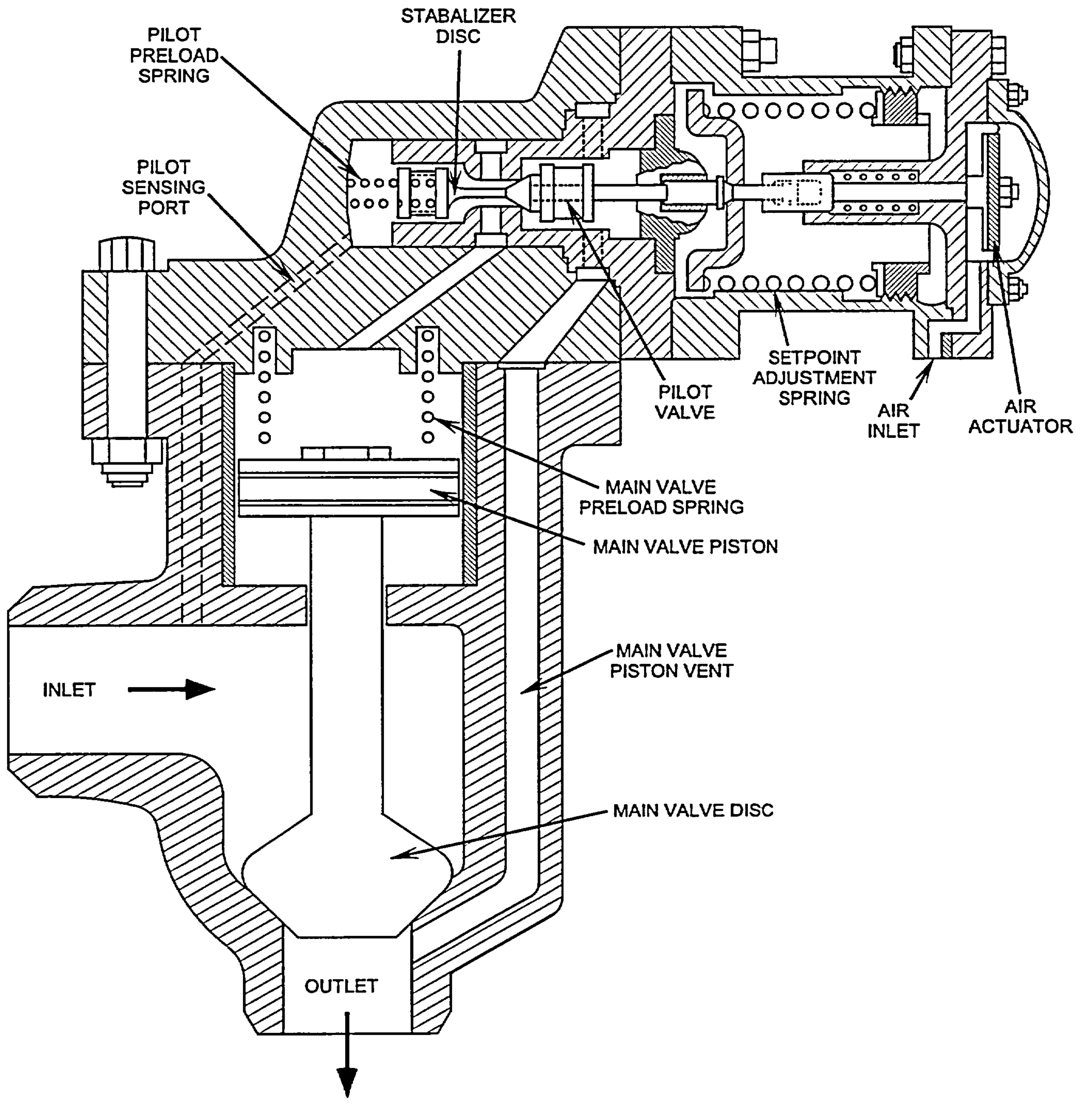

Figure 2.5-2 Two Stage Target Rock Safety/Relief Valve

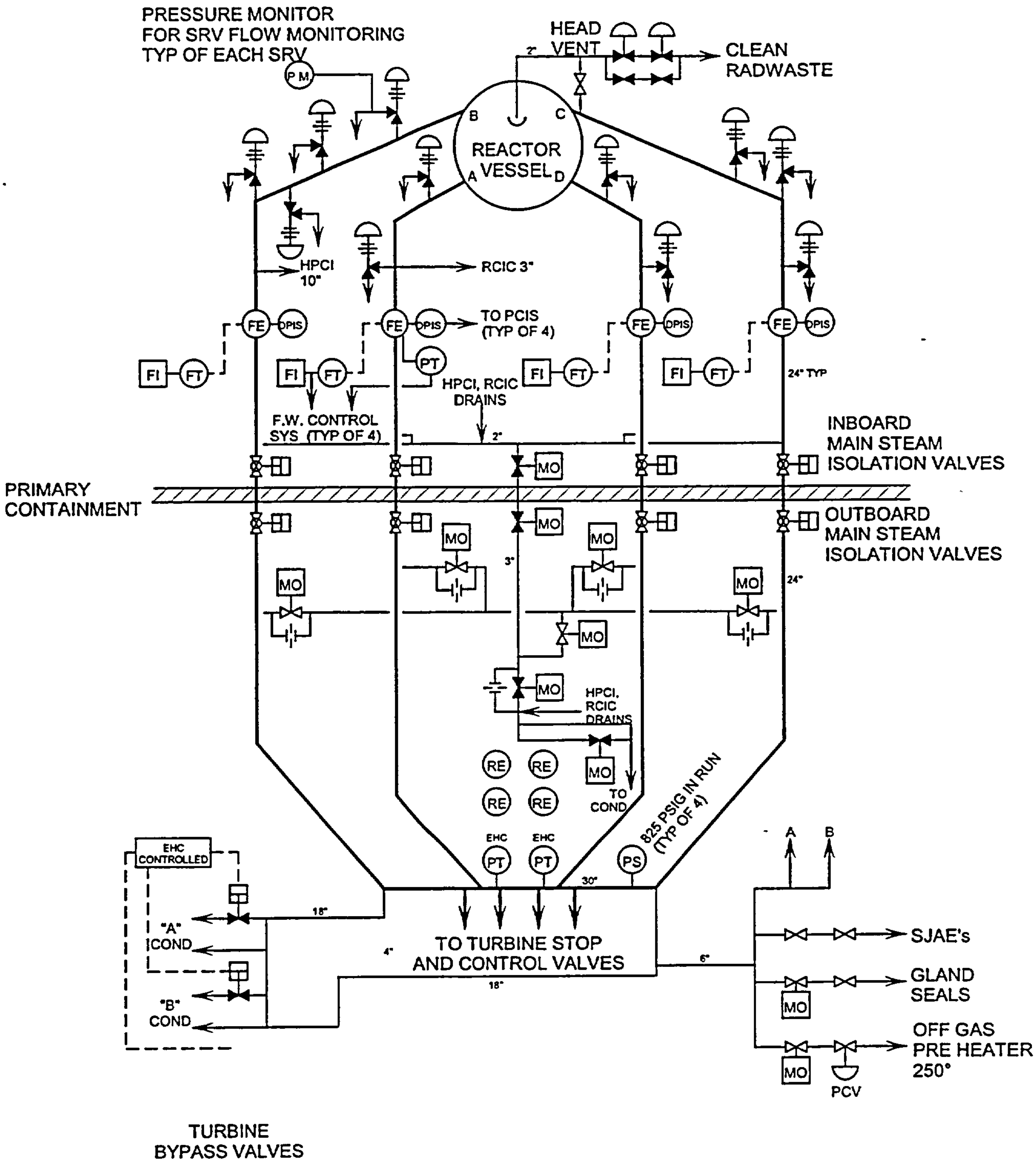

Figure 2.5-3 Main Steam System

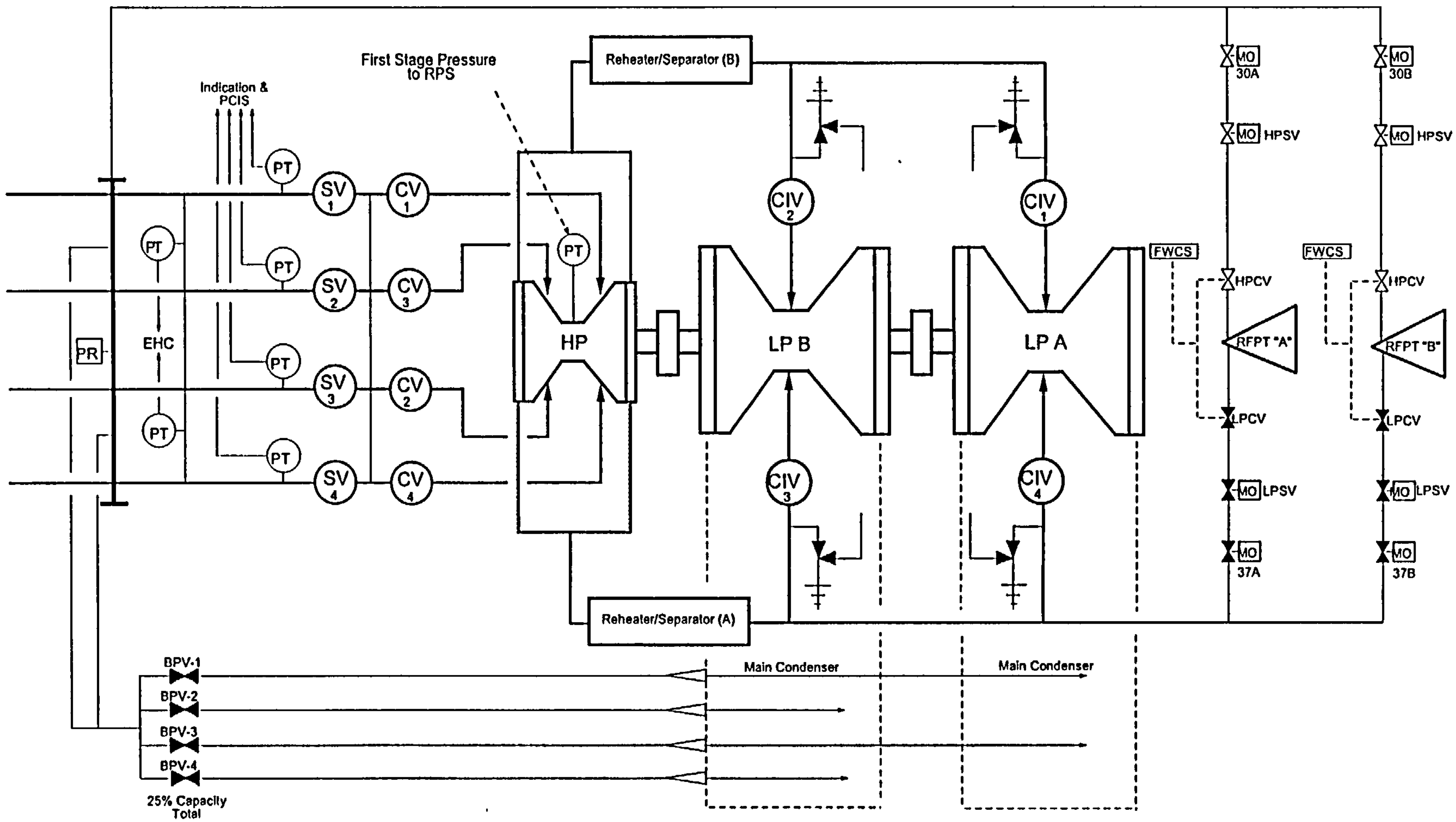

Figure 2.5-4 Main Steam System

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2.6 CONDENSATE AND FEEDWATER SYSTEM

The purposes of the Condensate and Feedwater System are:

1. To condense steam and collect drains in the main condenser.

2. To purify, preheat, and pump water from the main condenser to the reactor vessel.

The functional classification of the Condensate and Feedwater System is that of a power generation system.

2.6.1 System Description

The Condensate and Feedwater System, shown in Figure 2.6-1, is an integral part of the plant's conventional regenerative steam cycle. The steam exhausted from the three low pressure turbines is condensed in the main condenser and collected in the condenser hotwell, along with various equipment drains. The condensate that is collected in the hotwell is removed by three condensate pumps. The condensate pumps provide the driving force for the condensate which flows through the steam jet air ejector (SJAE) condensers, steam packing exhaust condenser, and offgas condenser performing a heat removal function. At this point the condensate is directed to the condensate filter/demineralizers and, through the process of filtration and ion exchange, impurities are removed. After the demineralizers, booster pumps are used to maintain the driving force of the condensate flow through strings of low pressure feedwater heaters. The feedwater pumps then take the condensate flow and further increase the pressure to a value above reactor pressure. The amount of feedwater flowing to the reactor vessel is controlled by varying the speed of the turbine driven reactor feed pumps. The discharge of the feedwater pumps is directed to the high pressure feedwater heater strings for the final stage of feedwater heating. Two feedwater lines penetrate the primary containment and then further divide into a total of six penetrations which enter the reactor vessel with each supplying feedwater to a feedwater sparger. The feedwater spargers distribute the flow of feedwater within the vessel annulus area.

2.6.2 Component Description

The components that comprise the Condensate and Feedwater System are discussed in the following paragraphs and illustrated in Figure 2.6-1.

2.6.2.1 Main Condenser

The main condenser is a surface condenser with three separate shells with each shell operating at the same vacuum. The condensers serve as the main heat sink for low pressure turbine exhaust and several other flows, such as exhaust steam from the reactor feed pump turbines, cascading low pressure heater drains, SJAE drains, turbine bypass valves, etc. The condenser also provides an area for noncondensible gases from the primary steam to be collected and exhausted via the SJAE's.

The condensed steam and various drains accumulate in the bottom of the condenser in an area termed the hotwell and are designated condensate water.

2.6.2.2 Condensate Pumps

Two motor driven condensate pumps, with a capacity of 50% each, remove water from the main condenser hotwell and discharge to parallel string of auxiliary condensers. The auxiliary condensers consist of two Steam Jet Air Ejector (SJAE) condensers, two offgas condensers, and one steam seal condenser.

2.6.2.3 Auxiliary Condensers

The auxiliary condensers are shell and tube type heat exchangers with condensate passing through the tubes condensing steam outside the tubes. The auxiliary condensers perform a dual function by condensing the steam and adding heat to the condensate system.

The Steam Jet Air Ejector (SJAE) condensers are used to condense the steam exhausted from the first stage SJAE (Section 8.1).

The purpose of the steam seal condenser is to condense the leak off from the main and reactor feed pump turbine seals. A vacuum is maintained in the condenser by one of two blowers which remove noncondensible gases and discharge through a 1.75 minute holdup to the plant stack (Section 8.1).

The offgas condenser removes excessive moisture from the offgas process mixture following the recombination of hydrogen and oxygen in the offgas system (Section 8.1).

2.6.2.4 Condensate Filter/Demineralizers

The purpose of the condensate filter/demineralizers is to treat the full flow of condensate from the condenser hotwell by removing impurities through filtration and ion exchange. Impurities consist of but are not limited to corrosion products and any circulating water in leakage to the condenser.
The condensate filter/demineralizers consist of nine filter/demineralizer units located in individual concrete cells and a common backwash and precoat system.

Each filter/demineralizer unit consists of a vertical cylindrical vessel which houses several nylon filter elements. The filter elements are coated with a mixture of powdered cation and anion exchange resins in hydrogen and hydroxyl forms. The coating is applied to the elements to a thickness of about 0.25 inches.

When a filter/demineralizer unit exceeds a specified differential pressure or when the effluent conductivity increases above 1μmho/cm the unit is removed from service. With the unit out of service, the spent resin is removed by reversing the normal flow path (backwashing) and sent to the Solid Radwaste System for disposal. A fresh resin slurry is then applied by the precoat system and the unit is placed back in service or standby until needed. The entire backwashing and precoat operation for a single unit requires about one hour.

2.6.2.5 Condensate Booster Pumps

Two, 50% capacity condensate booster pumps take suction on the filter/demineralizers outlet header and discharge to the low pressure feedwater heaters. The condensate booster pumps are motor driven with a discharge pressure of 600 psig.

2.6.2.6 Low Pressure Feedwater Heaters

The low pressure feedwater heaters are arranged in three parallel strings with each string consisting of one drain cooler and three low pressure heaters. The low pressure feedwater heaters utilize extraction steam from the three low pressure turbines to heat the feedwater prior to returning it to the reactor vessel thus increasing plant efficiency. The heaters are constructed so that condensate flows through the heater tubes while extraction steam flows around the tubes. The condensed extraction steam (extraction drains) from each heater flows to the next lower pressure heater by means of the pressure difference between the successive heaters.

2.6.2.7 Reactor Feedwater Pumps

The two feedwater pumps are 50% capacity pumps driven by variable speed steam turbines. The feedwater pumps take the heated feedwater from the outlet of the low pressure feedwater heaters and provide the driving force for returning the feedwater to the reactor vessel. The Feedwater Control System regulates the steam supply to the reactor feed pump turbines thus regulating the speed of the pump to produce more or less flow to the reactor vessel to control vessel water level.

Steam for the reactor feed pump turbines is supplied from two sources, main steam equalizing header and the high pressure turbine exhaust. During normal operation the steam from the high pressure turbine exhaust is used and the main steam equalizing header serves as the backup supply.

2.6.2.8 High Pressure Feedwater Heaters

The high pressure feedwater heaters represent the last stage of feedwater heating. They consists of three parallel strings with two heaters per string. The term high pressure heater is used because they use higher pressure extraction steam and are located at discharge of the feedwater pumps.

2.6.2.9 Feedwater Spargers

The outlet of the high pressure feedwater heaters is arranged so that two feedwater lines penetrate the primary containment. Containment isolation is provided by check valves on both sides of the drywell. Once inside the primary containment, the feedwater lines are divided into six lines. The six feedwater lines have separate reactor vessel penetrations and are equipped with thermal sleeves and a feedwater sparger inside the reactor vessel. The six feedwater spargers provide flow distribution in the vessel downcomer annulus area.

2.6.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

2.6.3.1 Startup Operation

During plant startup, feedwater is required to maintain water level in the reactor when the steaming rate exceeds the Control Rod Drive System (Section 2.3) input. To accomplish the task of maintaining reactor water level during plant startups, below the steam pressure used to roll the reactor feed pumps, the condensate and condensate booster pumps along with a startup bypass valve are used. The condensate and condensate booster pumps supply the motive force while the startup bypass valve, bypassing the feed pumps, is used to regulate the amount of water addition to the reactor vessel.

When reactor pressure reaches approximately 300 psig, the reactor feed pumps can be started.

2.6.3.2 Normal Operation

The number of condensate pumps, condensate booster pumps, and reactor feed pumps inservice depends upon the plant power level. Normally at 100% power all the condensate, condensate booster, reactor feed pumps and eight of the nine filter/demineralizers are in operation. At lower power levels, various combinations of pumps and filter/demineralizers are used such that the pumps and filter/demineralizers are operating under their design flow rate. In general the flow requirements are controlled by the Feedwater Control System (Section 3.3) regulating the speed of the feed pumps.

2.6.4 System Interfaces

The interfaces that this system has with other plant systems are discussed in the paragraphs which follow.

2.6.4.1 Feedwater Control System (Section 3.3)

The Feedwater Control System uses total feedwater flow as one of its controlling inputs in the three element logic. The FWCS also controls the speed of the RFP turbines and position of the feed pump bypass valve.

2.6.4.2 Reactor Water Cleanup System (Section 2.8)

The Reactor Water Cleanup System returns water to the reactor vessel via the feedwater lines.

2.6.4.3 Reactor Core Isolation CoolingSystem (Section 2.7)

The Reactor Core Isolation Cooling System returns water to the reactor vessel via the feedwater lines.

2.6.4.4 Main Steam System (Section 2.5)

The Main Steam System supplies steam to the RFPT's and feedwater heaters.

2.6.4.5 Offgas System (Section 8.1)

The Offgas System uses condensate flow as cooling water to the SJAE condensers, steam seal condenser, and offgas condenser.

2.6.4.6 High Pressure Coolant Injection System (Section 10.1)

The High Pressure Coolant Injection System returns water to the reactor vessel via the feedwater lines.

2.6.5 Summary

Classification:

Power generation system

Purpose:

To condense steam and collect drains in the main condenser. To purify, preheat, and pump water from the main condenser to the reactor vessel.

Components:

Condenser; condensate pumps; aux condensers; filter/demineralizers; condensate booster pumps; feedwater heaters; feed pumps.

System Interfaces:

Feedwater Control System; Control Rod Drive System; Reactor Water Cleanup System; Reactor Core Isolation Cooling System; High Pressure Coolant Injection System; Main Steam System; Offgas System.

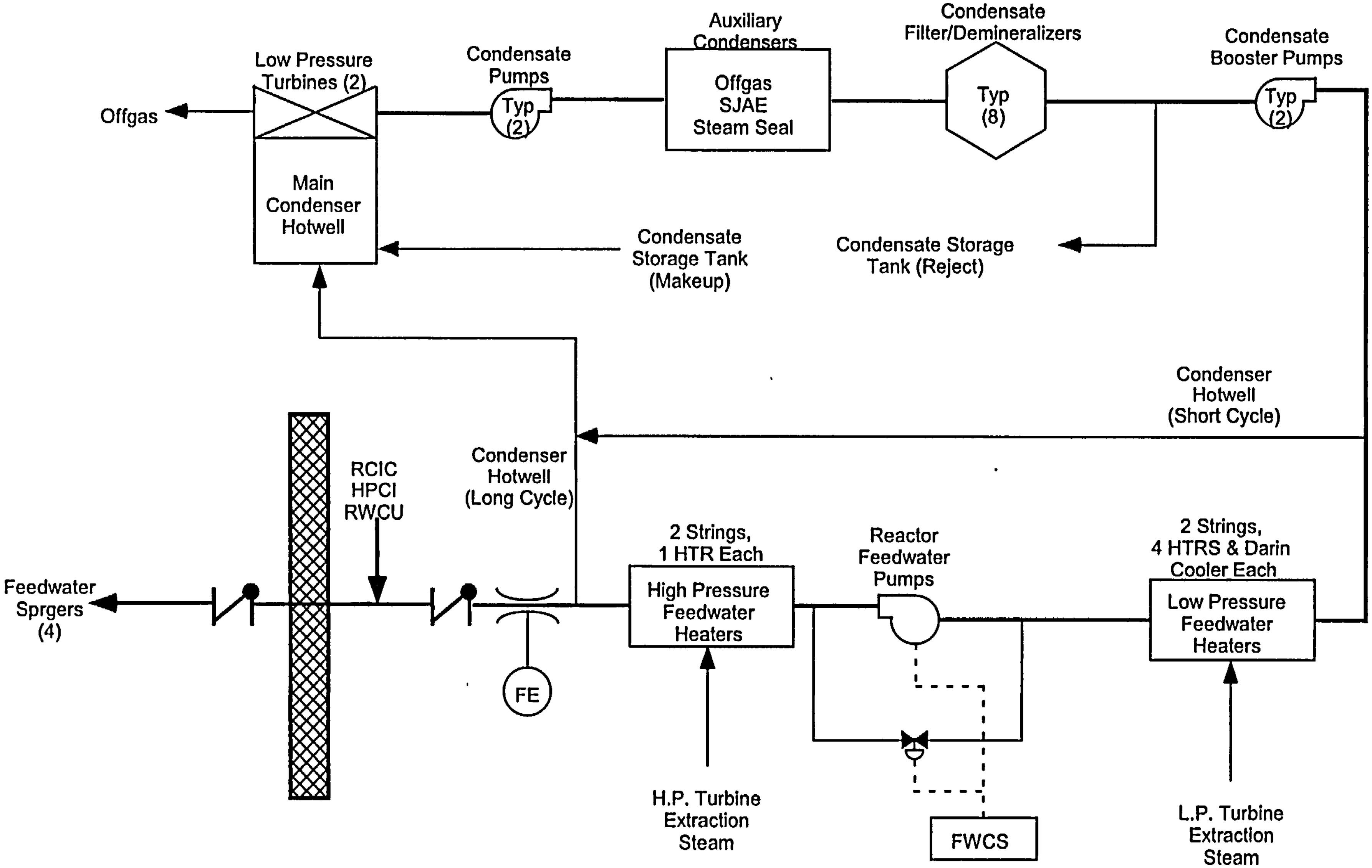

Figure 2.6-1 Basic Condensate and Feedwater System Drawing

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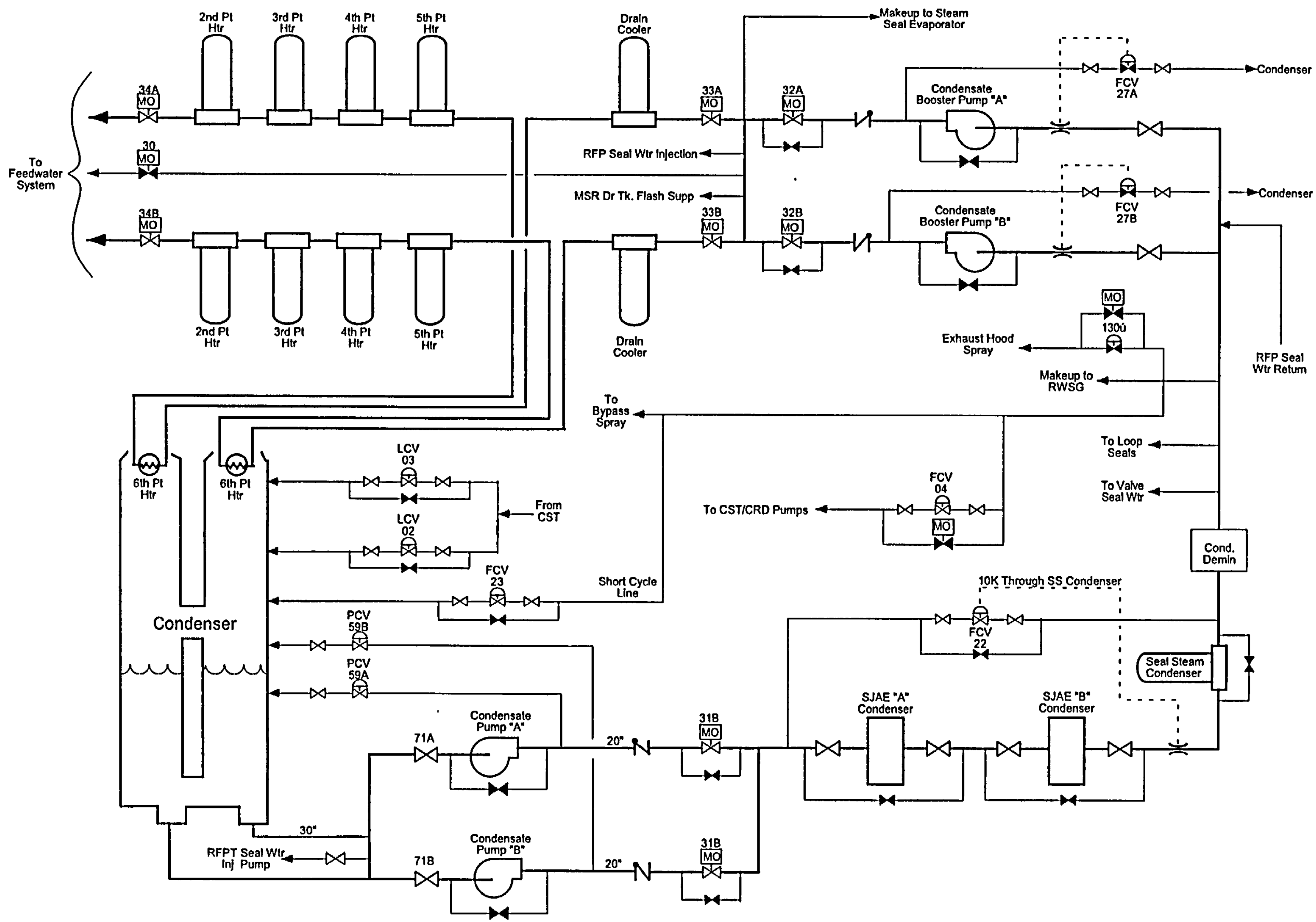

Figure 2.6-2 Condensate System

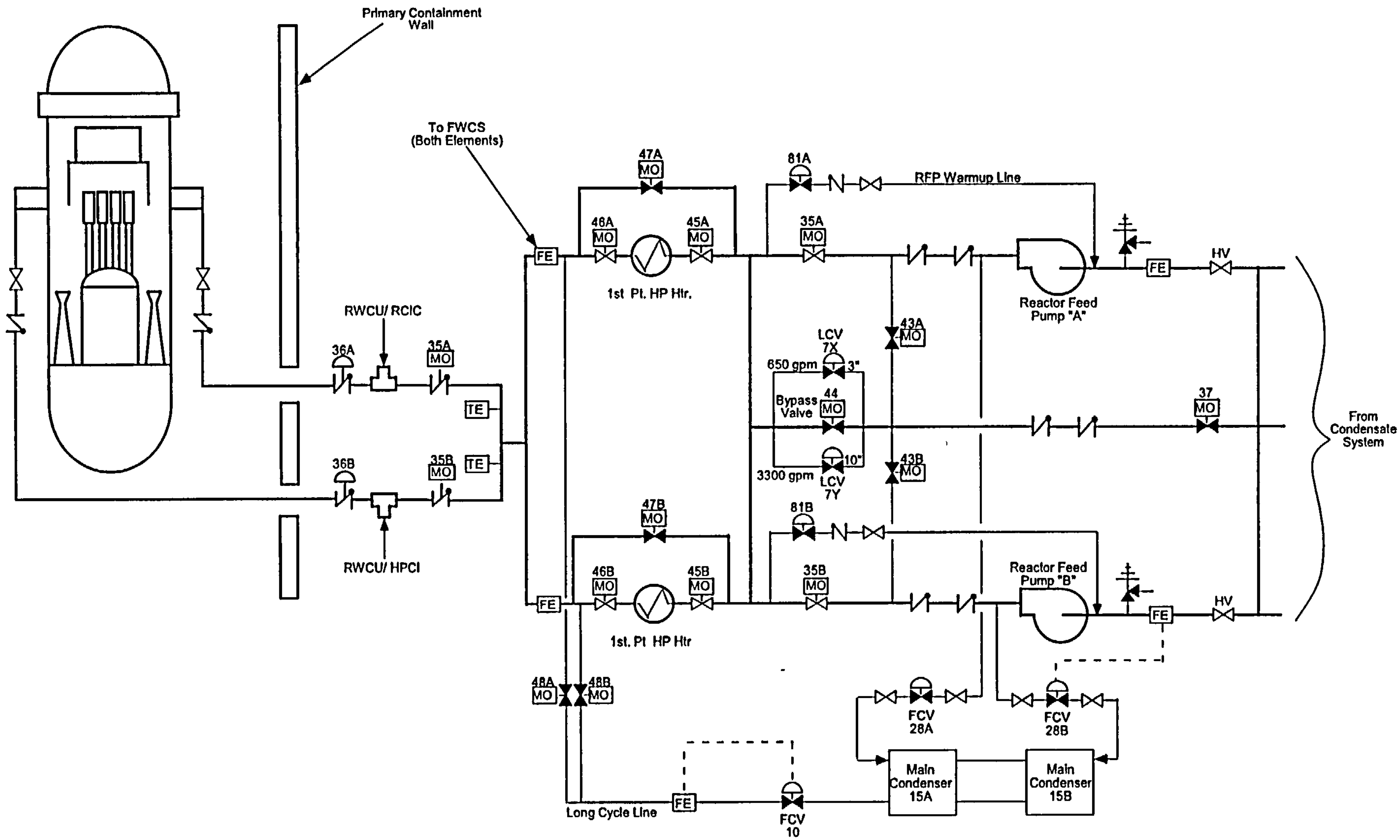

Figure 2.6-3 Feedwater System

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2.7 REACTOR CORE ISOLATION COOLING SYSTEM

The purposes of the Reactor Core Isolation Cooling (RCIC) System are to provide makeup water to the reactor vessel for core cooling when:

1. The main steam lines are isolated.

2. The condensate and feedwater system is not available.

The functional classification of the RCIC System is that of a safety related system.

2.7.1 System Description

The RCIC System (Figure 2.7-1) consist of a steam turbine driven pump, with the necessary piping and valves capable of delivering water to the reactor vessel at operating conditions.

The turbine is driven by steam produced in the reactor and exhausts to the suppression pool. The turbine driven pump unit supplies makeup water from the condensate storage tank, with an alternate supply from the suppression pool, to the reactor vessel via the feedwater piping. Additional discharge flow paths are provided to allow recirculation to the condensate storage tank for system testing and a pump minimum flow line to the suppression pool for pump protection. Sufficient capacity is provided to prevent reactor vessel level from decreasing below the top of the core. The system flow rate is approximately equal to reactor water boil off rate 15 minutes after shutdown with design decay heat.

Initiation of the RCIC System is accomplished automatically on reactor vessel water level low - low or manually by the operator.

2.7.2 Component Description

The components of the RCIC System are discussed in the paragraphs that follow.

2.7.2.1 Turbine Trip Throttle Valve

The RCIC turbine trip throttle valve is located upstream of the turbine control valve and provides for rapid turbine isolation during potentially unsafe turbine conditions. Opening of the valve, after a trip, is accomplished from the control room via a control switch. Closure of the valve is by spring force and opening is done by an electric motor.

2.7.2.2 Turbine Control Valve

The turbine control valve is opened by spring force and closed by oil pressure. RCIC turbine speed, and therefore pump flow, can be controlled by throttling the steam supply to the turbine. The RCIC flow control circuit determines the valve position necessary to produce the required pump discharge flow and then adjusts the oil pressure to the control valves.

2.7.2.3 RCIC Turbine

The RCIC turbine is designed to accelerate rapidly from a cold standby condition to full load conditions within 20 seconds. The RCIC turbine is a horizontally mounted radial re-entry, noncondensing turbine designed to operate with a steam supply pressure ranging from 1150 psig to 150 psig. The RCIC turbine operates with a 10 psig exhaust pressure, exhausting to the suppression pool under water.

2.7.2.4 Pump Suction Valves

Motor operated valves are used to line up RCIC pump suction from either the condensate storage tank or from the suppression pool.

Check valves are provided to prevent flow between the condensate storage tank and the suppression pool when both suction valves are open. Any time level in the condensate storage tank reaches a minimum level or the suppression pool reaches a maximum level, the suppression pool suction valve opens. When the suppression pool suction valve is fully open, a close signal is transmitted to the condensate storage tank suction and test return valves.

2.7.2.5 RCIC Pump

The RCIC pump is a turbine driven, horizontal, multistage, centrifugal pump with a capacity of 400 gpm. The RCIC pump is designed for water temperature between 40°F and 140°F. To ensure adequate net positive suction head (NPSH), the pump is located at a lower elevation than its sources of water.

2.7.2.6 RCIC Flow Controller

The RCIC system utilizes a flow controller to automatically or manually control system flow upon initiation. Selection of either automatic or manual mode is performed by the control room operator. In the automatic mode (normal position), the controller compares actual RCIC System flow (sensed by a flow element on the discharge of the pump) with the desired flow setpoint (adjusted by the operator at the controller). Any deviation between actual and desired flow is then converted into a hydraulic signal which positions the control valves as required to balance the flow signals. In manual mode the operator has direct control of RCIC system flow. The operator simply adjusts a manual potentiometer, at the flow controller, to create a signal used for positioning the control valves to obtain the desired flow.

2.7.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

2.7.3.1 RCIC Automatic Initiation

Following a reactor scram from power operation, fission product decay continues to produce heat. This heat will raise reactor pressure causing steam flow to the bypass valves, or through safety/relief valves. The steam flow will cause vessel water level to decrease if the feedwater system is not available. The RCIC system will maintain sufficient reactor water level until the reactor is depressurized and cooled down to a point where the shutdown cooling mode of the Residual Heat Removal System (Section 10.4) can be placed in operation.

The RCIC System is automatically initiated if vessel level decreases to a low - low water level. When the initiation signal is received, several actions occur. The turbine steam supply shutoff valve opens to admit steam to the turbine. The normal discharge path to the reactor is aligned, the test return line isolates and at the same time, the discharge valve to the feedwater piping opens.

As the turbine begins to roll, an attached lube oil pump builds up oil pressure and the turbine flow control system begins to throttle the turbine control valve. As the turbine speed continues to increase pump flow and discharge pressure increase until a flow of 400 gpm is achieved (flow controller setting).

Once initiated the RCIC will remain in operation until manually secured an automatic isolation signal is generated, a turbine trip signal is received, or reactor vessel water inventory reaches +56" (high level point). At +56" increasing, the turbine steam supply shutoff valve closes to secure the RCIC System. The system will automatically reinitiate when level decreases to the low - low setpoint.

2.7.3.2 Automatic Isolation

The RCIC System steam supply line is an integral part of the nuclear system process barrier and it will be automatically isolated upon detection of a leak. Automatic isolation signals include any one of the following: RCIC high steam line flow, RCIC steam line area temperature high, low steam supply pressure, or high pressure between turbine exhaust rupture diaphragms.

Once an isolation signal is generated, the following automatic actions occur:

1. Isolation signal seals in.
2. Inboard and outboard steam supply isolation valves shut.
3. Turbine trip throttle valve trips closed.
4. Pump discharge valve shuts.
5. Pump minimum flow valve shuts if open.

2.7.3.3 RCIC Turbine Trips

The RCIC turbine is automatically shut down by closing the turbine trip throttle valve to protect the physical integrity of the RCIC System. If any of the following conditions are detected the RCIC System will automatically trip:

1. Isolation signal.
2. High turbine exhaust pressure.
3. Low pump suction pressure.
4. Overspeed.

2.7.4 System Interfaces

A short discussion of interfaces this system has with other plant systems is given in the paragraphs which follow.

2.7.4.1 Main Steam System (Section 2.5)

The RCIC System steam line taps off one of the main steam lines upstream of the inboard main steam isolation valves.

2.7.4.2 Primary Containment System (Section 4.1)

The suppression pool is the alternate source of water for the RCIC pump. It also condenses the RCIC turbine exhaust steam. RCIC System minimum flow water is routed to the suppression pool.

2.7.4.3 Condensate and Feedwater System (Section 2.6)

The RCIC System flow is delivered to the vessel via the Condensate and Feedwater System. Additionally, the steam supply line upstream of the RCIC steam supply shutoff valve is normally kept pressurized with steam. Condensate which forms in this piping collects in the drain pot and flows to the main condenser through a drain trap. The drain trap is isolated during RCIC turbine operation. Water is normally supplied to the system from the condensate storage tank.

2.7.5 Summary

Classification :

Safety related system

Purpose:

To provide makeup water to the reactor vessel for core cooling when: The main steam lines are isolated. The condensate and feedwater system is not available.

Components:

Steam supply isolation valves; turbine trip throttle valve; turbine control valve; turbine; turbine driven pump; and flow controller.

System Interfaces:

Main Steam System; Primary Containment System; Condensate and FeedwaterSystem.

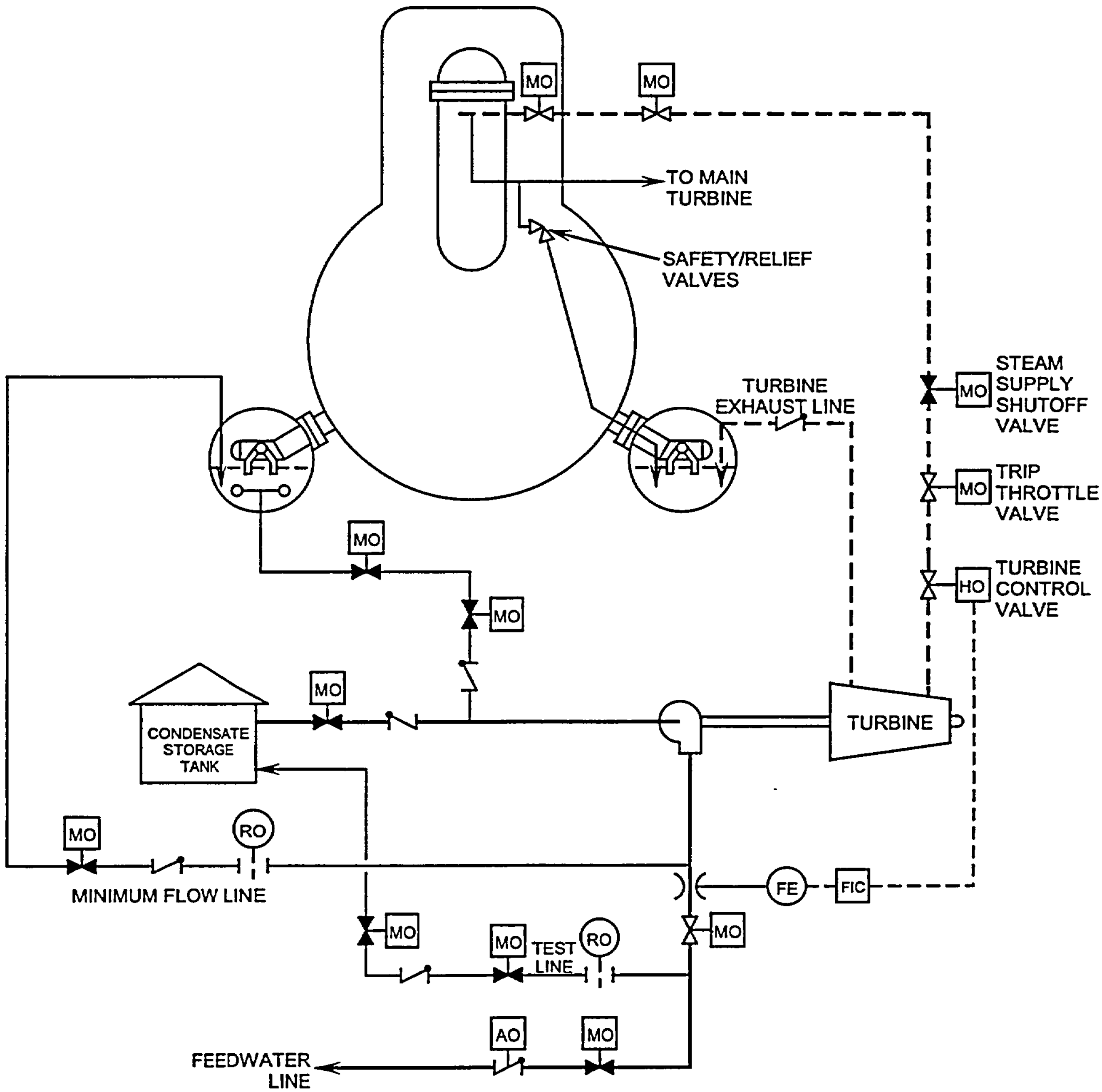

Figure 2.7-1 Reactor Core Isolation Cooling System

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2.8 REACTOR WATER CLEANUP SYSTEM

The purposes of the Reactor Water Cleanup (RWCU) System are:

1. To maintain reactor water quality by filtration and ion exchange.

2. To provide a path for removal of reactor coolant when required.

The functional classification of the RWCU System is that of a power generation system.

2.8.1 System Description

The RWCU System, shown in Figure 2.8-1, provides a continuous purification of a portion of the reactor recirculation system flow. Water is taken from the Reactor Recirculation System (see section 2.4) piping and pumped through the regenerative and nonregenerative heat exchangers, filter/demineralizers and returned through the shell side of the regenerative heat exchanger to the feedwater piping. The RWCU system also removes a small portion of water from the reactor vessel bottom head drain to reduce any crud buildup and possibility of cold water stratification.

2.8.2 Component Description

The major components of the RWCU system are discussed in the paragraphs which follow and can be seen in Figure 2.8-1.

2.8.2.1 RWCU Inlet Piping

The major flow of the RWCU system is taken from the suction of the 'A' recirculation pump. This suction is common to the shutdown cooling mode of the Residual Heat Removal System (Section 10.4). An additional source of water is provided by a line from the reactor vessel bottom head drain. Isolation valves are provided both inside and outside of the primary containment for isolating the RWCU system during unsafe conditions.

2.8.2.2 RWCU Pumps

Two RWCU pumps provide the needed motive force for water flow from the recirculation loop, through the RWCU system and back to the reactor vessel via the feedwater system. The RWCU pumps are motor driven, horizontally mounted, centrifugal pumps. Both pumps together provide 100% system flow which corresponds to approximately 1% of rated feedwater flow.

2.8.2.3 Regenerative Heat Exchangers

The first stage of temperature reduction is accomplished by three regenerative heat exchangers arranged in series. The regenerative heat exchangers transfer heat from the influent reactor water by preheating the RWCU system's effluent. This results in less system heat loss and less thermal stress as the effluent water enters the feedwater system.

2.8.2.4 Nonregenerative Heat Exchangers

The nonregenerative heat exchangers provide the final reduction of water temperature prior to entry into the filter/demineralizers. Unlike the regenerative heat exchangers ;the heat removed in the nonregenerative heat exchangers is lost to another system, the Reactor Building Closed Cooling Water System (Section 11.1).

2.8.2.5 Filter/Demineralizer Units

The purpose of the filter/demineralizers is to remove soluble and insoluble impurities from the reactor water. This process is accomplished by the water flowing through a finely ground filter aid material (Solka-Floc) and ion exchange resins.

The two filter/demineralizer units (Figure 2.8-2) are of a pressure precoat type in which the vessel filter tubes are coated with a filter aid and powdered ion exchange resin. These materials serve as the filter media and ion exchange agent and are held on to the screen type tubes by the differential pressure generated by the water flowing through each unit. Flow through the system is automatically controlled by flow control valves located at the outlet of each filter/demineralizer.

Each unit is composed of a pressure vessel which contains the filter tubes, a holding pump, instrumentation, controls, valves, and piping. Air operated inlet and outlet isolation valves and the flow control valve are furnished for each filter/demineralizer unit to permit individual operation and serving. Auxiliary components required for precoating, backwashing, and disposal of the filter media are shared by both filter/demineralizers.

2.8.2.6 RWCU Outlet Piping

Filter/demineralizer effluent line branches into two lines, one line routing water to the regenerative heat exchangers and the other to the blowdown control section. During normal system operation, flow is routed through the shell side of the regenerative heat exchangers, a motor operated discharge isolation valve and into the feedwater lines.

During other operations requiring reactor vessel blowdown, a portion of the flow is routed to the main condenser (Section 2.6) or to the Liquid Radwaste System (Section 8.2). Blowdown flow is directed through a common header which is controlled by a flow control valve and a restricting orifice. The restricting orifice provides a large pressure drop to protect the low pressure downstream piping. If additional flow is required, a motor operated orifice bypass valve can be opened.

2.8.3 System Features

A short discussion of the system features is given in the paragraphs which follow.

2.8.3.1 System Operation

The RWCU system is normally operated continuously during all phases of reactor plant operation to maintain undissolved solids less than 0.01 ppm and conductivity less than 0.1 micro mho/cm. Blowdown (draining) is normally used only to remove excess water from the reactor vessel.

Reactor water quality depends on several factors: corrosion of primary system materials, input of impurities and corrosion products from the Condensate and Feedwater System, piping deposition and re-entrainment at system surfaces, radioactive decay, and removal of impurities (corrosion products and fission products) by the filter/demineralizers.

Impurities in the reactor water are of two forms, soluble and insoluble. By definition, insoluble material is that collectible on a type HA millipore filter paper; the remainder is soluble. The soluble materials are suspended in the water until collected or plated onto some surface.

The insoluble materials are removed by filtration. Corrosion product inputs from the primary systems have been minimized by use of stainless steel vessels, pipes, and heat exchangers. The majority of corrosion product input originates in the condenser and feedwater components where carbon steel piping and components are used. Cleanup system operation is necessary to maintain reactor water purity. Reactor operation without the cleanup system is limited to relatively short periods of time.

2.8.3.1.1 Normal Operation

During normal operation water is taken from the reactor recirculation system piping and pumped through the regenerative and nonregenerative heat exchangers, both filter/demineralizers, and returned through the shell side of the regenerative heat exchanger to the feedwater piping. The RWCU also removes a small portion of water from the reactor vessel bottom head drain (Section 2.1) to reduce any crud buildup.

2.8.3.1.2 Blowdown

During a plant startup, shutdown, or low steaming rates, it is sometimes necessary for the operator to drain water from the vessel in order to maintain a proper vessel level. This is done with the blowdown portion of the RWCU System. Water, from the outlet of the filter/demineralizers, can be rejected to the main condenser or to the Liquid Radwaste System rather than being returned to the vessel. In this way water is removed from the reactor. The blowdown flow rate is controlled manually from the control room as shown on Figure 2.8-2. The hand operated controller is set to the desired flow and the appropriate motor operated valve is opened, directing flow to either the main condenser or to the Liquid Radwaste System. The main condenser is the preferred blowdown point in order to limit the duty on the liquid waste processing facilities. During the blowdown operation, the greatest duty is imposed on the non-regenerative heat exchangers because much of the return flow that was going to the regenerative heat exchangers is now bypassing them. If, during blowdown operation, high pressure is sensed downstream of the blowdown control valve or if low pressure is sensed upstream, the blowdown control valve is automatically closed. The high pressure sensor protects the piping downstream of the control valve and the low pressure sensor prevents draining the entire RWCU System piping in a siphon action to the main condenser or radwaste.

2.8.3.2 Filter/Demineralizer Operations

Two filter/demineralizer operations are discussed in the paragraphs which follow.

2.8.3.2.1 Backwashing

As the differential pressure across a filter/demineralizer unit approaches 20 psid, or if the outlet conductivity reaches 0.1μmhos, the filter/demineralizer is removed from service for backwashing and precoating.

Backwashing consists of removing the spent resins from the filter/ demineralizer holding elements. This is accomplished by the use of an air blast injected into the filter/demineralizer to dislodge the precoat. Condensate is then pumped into the filter/ demineralizer through the outlet line. The drain line is then opened, and the mixture of water and spent resins is pumped to the cleanup phase separator tank of the radwaste facilities.

2.8.3.2.2 Precoating

After the filter/demineralizer is backwashed, it is now ready for a new application of resins. The filter/demineralizer is precoated by circulating a slurry of the resin from the resin feed tank onto the stainless steel resin holding elements.

The slurry deposits evenly on the elements while the water returns to the resin feed tank. Precoat recirculation continues until the return water is clear. At this time, a holding pump is automatically started to maintain the precoat in place on the holding elements. The filter/demineralizer is then returned to service, and the holding pump flow stops as system flow increases.

The holding pump for each filter/demineralizer unit is automatically restarted if the filter/demineralizer outlet flow drops below 90%. This ensures a flow through the filter/demineralizer at all times and prevents the precoat from falling off the elements.

2.8.3.3 Isolation

The inboard and outboard system supply isolation valves automatically close following an isolation signal from any one of the following:

1. Low reactor water level
2. Standby Liquid Control System actuation
3. High temperature filter/demineralizer inlet
4. High area temperature
5. Area drain temperature high

2.8.4 System Interfaces

A short discussion of interfaces this system has with other plant systems is given in the paragraphs which follow.

2.8.4.1 Recirculation System (Section 2.4)

The Recirculation System supplies water to the RWCU System.

2.8.4.2 Reactor Vessel System (Section 2.1)

The reactor vessel provides water to the RWCU System via the bottom head drain line.

2.8.4.3 Condensate and Feedwater System (Section 2.6)

The Condensate and Feedwater System provides a return path to the vessel for the processed water.

The main condenser can be used to collect water from RWCU System blowdown.

2.8.4.4 Liquid and Solid Radwaste Systems (Sections 8.2 and 8.3)

The radwaste facilities are used to collect water from RWCU blowdown and spent resins from the filter demineralizers.

2.8.4.5 Reactor Building Closed Cooling Water System (Section 11.1)

The Reactor Building Closed Cooling Water System provides cooling water to the RWCU pumps and the non regenerative heat exchangers.

2.8.5 Summary

Classification :

Power Generation System

Purpose:

To maintain reactor water quality by filtration and ion exchange.Provides a path for removal of reactor coolant when required.

Component:

Isolation valves; pumps; regenerative heat exchangers; nonregenerative heat exchangers; filter/demineralizers.

System Interfaces:

Recirculation System; Reactor Vessel and Internals System; Condensate and Feedwater System; Radwaste Systems; Reactor Building Closed Cooling Water System

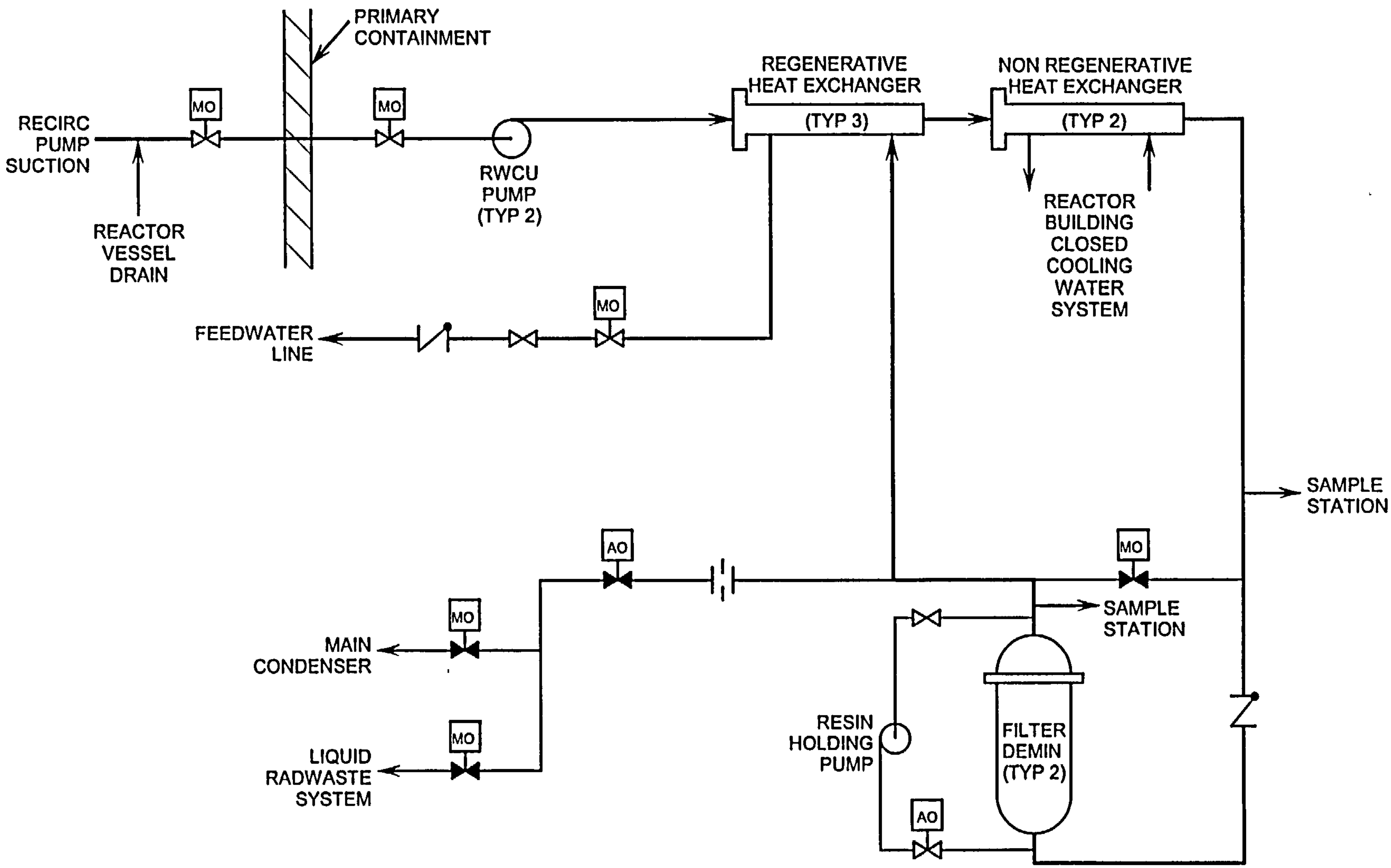

Figure 2.8-1 Reactor Water Cleanup System

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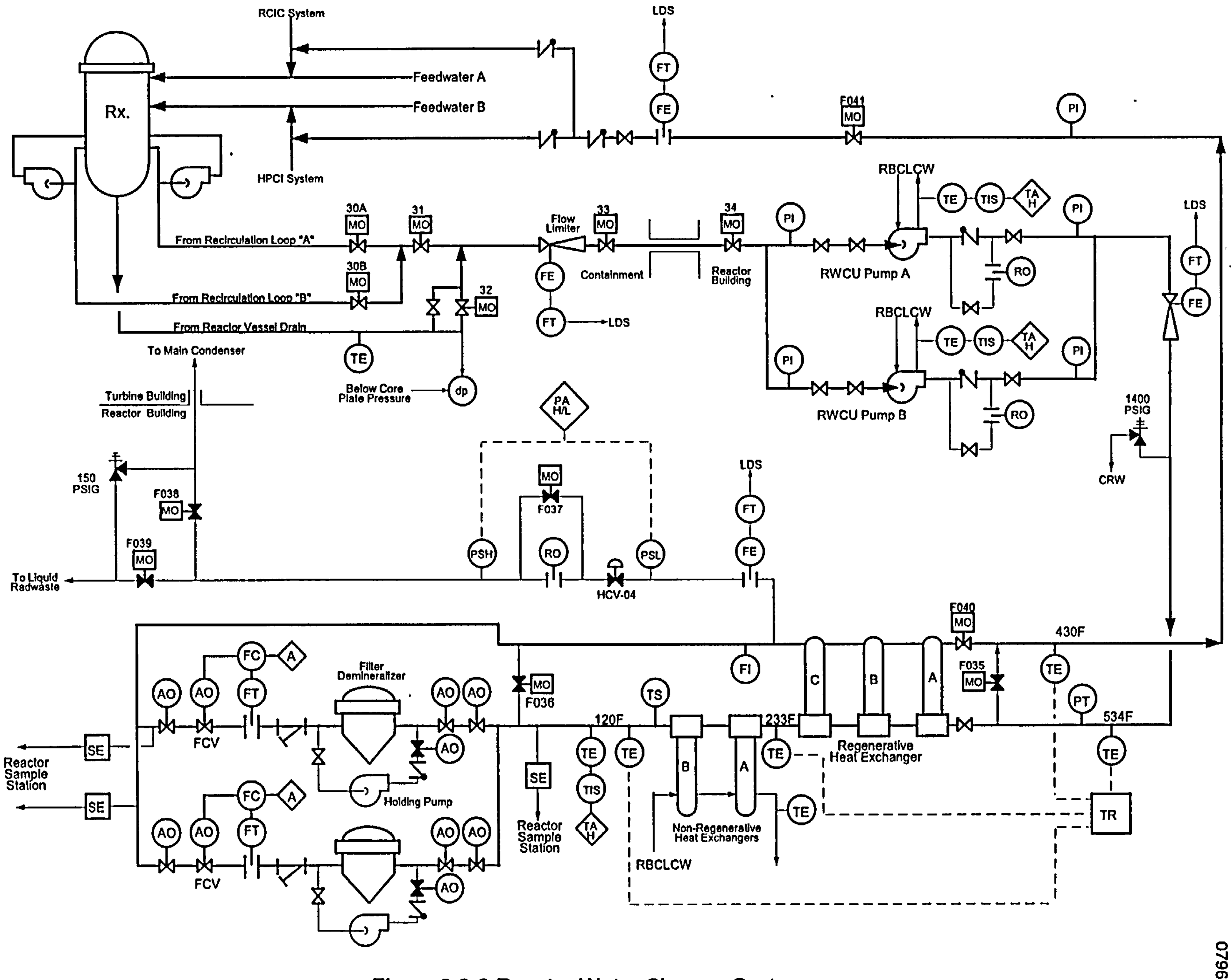

Figure 2.8-2 Reactor Water Cleanup System